



Atomic Energy
Control Board

P.O. Box 1046
Ottawa, Canada
K1P 5S9

Commission de contrôle
de l'énergie atomique

C.P. 1046
Ottawa, Canada
K1P 5S9

INFO-0323

**REJECTION INDEX
FOR PRESSURE TUBES**

by

A.B. Mitchell

and

D. Meneley

A report prepared for the
Atomic Energy Control Board
Ottawa, Canada

Project Number 2.150.1

October 1989

Canada

Research report



ABSTRACT

The objective of the present study was to establish a set of criteria (or "Rejection Index") which could be used to decide whether a zirconium-2 1/2 w/o niobium pressure tube in a CANDU reactor should be removed from service due to in-service degradation.

Information available in the AECB library up to November 1988 was used to review the following topics:

- pressure tube failure events in Canada and worldwide;
- metallurgical degradation of zirconium alloy pressure tubes in reactor service;
- "leak-before-break" and other safety philosophies;
- pre-service and in-service inspection practices, and
- current requirements for periodic inspection of pressure tubes.

On the basis of this background information, a critique of key issues associated with establishing a realistic "rejection index" was prepared. Areas of uncertainty in available information were identified and recommendations for further analysis and laboratory testing made.

A "Rejection Index" based on the following limits has been recommended:

(a) Limits related to design intent and normal operation:

- any garter spring must remain within the tolerance band specified for its design location;
- the annulus gas system must normally be operated in a circulating mode with a procedure in place for purging to prevent accumulation of deuterium. It must remain sensitive to leaks into any part of the system, and
- pressure tube dimensions and distortions must be limited to maintain the fuel channels within the original design intent.

(b) Limits related to defect tolerance:

- adequate time margins between occurrence of a leaking crack and unstable failure must be demonstrated for all fuel channels;
- long lap-type flaws are unacceptable;
- crack-like defects of any size are unacceptable, and
- score marks, frat marks and other defects with contoured profiles must fall below certain depth, length and stress intensity limits.

(c) Limits related to property degradation:

- at operating temperature each pressure tube must be demonstrated to have a critical length in excess of a stipulated value;
- the maximum equivalent hydrogen level in any pressure tube should not exceed a limit which should be defined taking into account the known history of that tube;
- the maximum equivalent hydrogen level in any rolled joint should not exceed a limit which is presently recommended as 200 ppm equivalent hydrogen, and
- the maximum diametral creep strain should be limited to less than 5%.

RÉSUMÉ

La présente étude a pour but d'établir une série de critères (ou «index de retrait») à utiliser pour décider si un tube de force de réacteur CANDU, fait d'un alliage de zirconium et de niobium à 2,5 pour 100 en poids devrait être déclassé ou non par suite de dégradation en cours de fonctionnement.

Les auteurs ont utilisé la documentation qui se trouvait dans la bibliothèque de la CCEA en novembre 1988 pour étudier les questions suivantes :

- les cas de ruptures de tubes de force au Canada et dans le monde;
- la dégradation métallurgique des tubes de force en alliage de zirconium durant l'exploitation des réacteurs;
- le principe de la «fuite avant la rupture» et autres approches de sûreté;
- les pratiques d'inspection avant et durant le fonctionnement;
- les exigences actuelles d'inspection périodique des tubes de force.

D'après la documentation disponible, les auteurs ont analysé les questions clés reliées à l'établissement d'un «index de retrait» réaliste. Ils ont déterminé les incertitudes de la documentation et fait des recommandations pour que d'autres analyses et essais en laboratoire soient effectués.

Ils ont ensuite recommandé un «index de retrait» à partir des limites suivantes :

a) Limites reliées aux spécifications nominales et à l'exploitation normale :

- tout ressort expandeur doit conserver l'espacement de tolérance prévu dans sa conception;
- le circuit de gaz annulaire doit normalement fonctionner en mode circulatoire et une procédure doit exister pour que le circuit se purge automatiquement afin d'éviter l'accumulation de deutérium. Le circuit doit rester sensible aux fuites possibles vers toute autre partie du système;
- les dimensions et les déformations des tubes de force doivent être limitées pour maintenir la conception originale des canaux de combustible.

b) Limites reliées aux défaillances tolérables :

- les délais de tolérance entre l'apparition d'une fissure et toute défaillance instable doivent être déterminés pour tous les canaux de combustible;
- les défauts de fabrication dus au sertissage sont inacceptables;
- les fissures de toutes tailles sont inacceptables;
- les entailles, les marques d'usure et les autres défauts à contours profilés doivent respecter certaines limites de profondeur, de longueur et de contrainte.

c) Limites reliées à la dégradation des propriétés :

- à la température de fonctionnement, on doit prouver que la longueur de fissure critique de chaque tube de force est supérieure à la valeur préétablie;
- le niveau d'équivalent de maximum d'hydrogène de chaque tube de force ne devrait pas dépasser la limite qui devrait être établie en tenant compte du rendement antérieur connu du tube visé;
- le niveau d'équivalent maximum d'hydrogène dans tout assemblage à manchon écroui ne devrait pas dépasser la limite recommandée de 200 ppm d'équivalent d'hydrogène;
- la contrainte maximale du fluage diamétral devrait être limité à 5 pour 100.

DISCLAIMER

The Atomic Energy Control Board is not responsible for the accuracy of the statements made or opinions expressed in this publication and neither the board nor the authors assume liability with respect to any damage or loss incurred as a result of the use made of the information contained in this publication.

This report was prepared by the New Brunswick Research and Productivity Council (RPC) and the University of New Brunswick (UNB) under contract to the Atomic Energy Control Board (AECB).

AECB Project No.: 2.150.1

Dept. of Supply & Services Serial No. 063SS-87055-8-4010EX00PT01

RPC Report No. EM/89/99

RPC Job No. 10000758

Date: March 31, 1989

ACRONYMS AND ABBREVIATIONS

ASME	American Society of Mechanical Engineers
BBL	Break-Before-Leak
BWR	Boiling Water Reactor
CANDU-PHW	Canadian Deuterium Uranium Pressurized Heavy Water
CCL	Critical Crack Length
COG	Candu Owners Group
CSA	Canadian Standards Association
DHC	Delayed Hydride Cracking
LBB	Leak-Before-Break
LOCA	Loss of Coolant Accident
NGS	Nuclear Generating Station
NPD(R)	Nuclear Power Demonstrator (Reactor)
PHW(R)	Pressurized Heavy Water (Reactor)
PRTR	Plutonium Recycle Test Reactor
SDS1	Shut Down System One
SGHWR	Steam Generating Heavy Water Reactor
SLAR	Spacer Location and Relocation
TSS	Terminal Solid Solubility
UKAEA	United Kingdom Atomic Energy Authority
USNRC	United States Nuclear Regulatory Commission
w/o or wt%	Percentage by Weight

TABLE OF CONTENTS

1.0 INTRODUCTION	1
1.1 General	1
2.0 REJECTION CRITERIA PHILOSOPHIES	6
2.1 Jurisdictional Requirements	6
2.2 Alternative Inspection Philosophies	9
2.3 Leak-Before-Break Concept (LBB)	11
2.4 Leak-Before-Break Applied to Pressure Tubes	14
3.0 PRESSURE TUBE FAILURE IN NORMAL OPERATION, UPSET AND ACCIDENT CONDITIONS	24
4.0 PRESSURE TUBE PERFORMANCE	27
5.0 POTENTIAL LIFE LIMITING DEGRADATIONS	30
5.1 General Considerations	30
5.2 Corrosion, Oxidation and Deuterium Pick-up	30
5.3 Fabrication Defects	35
5.4 Rolled Joint Pull-Out Strength	36
5.5 Fracture Toughness	36
5.6 Delayed Hydride Cracking	38
5.7 Blister Formation	42
5.8 Irradiation Induced Elongation	43
5.9 Irradiation Accelerated Diametral Creep	44
5.10 Irradiation Accelerated Sag	45
5.11 Fatigue Cracking	46
5.12 Construction and Operating Deficiencies	46
6.0 RECOMMENDED BASIS FOR REJECTION INDEX	48

7.0 SCHEMATIC INSPECTION PROGRAMS	51
8.0 GENERAL CONCLUSIONS	51
9.0 ACKNOWLEDGEMENTS	54
APPENDIX A BIBLIOGRAPHY OF INFORMATION	
APPENDIX B PRESSURE TUBE PERFORMANCE	
APPENDIX C SUMMARY OF AUTHOR'S RECOMENDATIONS FOR FURTHER WORK	

1.0 INTRODUCTION

1.1 General

For licensing purposes, each Candu-PHW reactor unit receives, at the design stage, a detailed safety analysis which includes a demonstration that failure in-service, of a major component such as a pressure tube, can be accommodated without undue hazard to the public. An implicit assumption within this study is the very low probability of metallurgical failure of components properly designed to the requirements of the ASME Section III Code.

As an additional safeguard, during subsequent plant operation, periodic inspection of selected samples of nuclear power plant components are carried out to the rules identified in CSA CAN3-N285.4 (hereinafter referred to as N285.4) in order to demonstrate in a practical manner, that the likelihood of component failures is low at the time the plant enters service and does not increase significantly due to progressive deterioration in-service. Periodic inspection is aimed at materials which are not expected to degrade significantly in-service.

To date, experience with Candu-PHW reactor components made from conventional constructional materials (e.g. carbon steels, stainless steels, nickel alloys, etc.) suggests that this approach is adequate. Significant deteriorations of these components within the Candu-PHW nuclear systems has been both relatively rare and by mechanisms predictable from the many years of worldwide experience accumulated for these materials. The main changes in rules governing these components introduced over the last few years have focused on more specific methods for measuring defect severity and the analysis required for assessing their rejectability.

In the case of Candu-PHW pressure tubes, both Zircaloy-2 and zirconium-2½% niobium alloys used have proved to be susceptible to the in-reactor environmental factors of neutron irradiation, corrosion and stress. Various forms of in-reactor degradation have been experienced since 1972 relating to dimensional

distortion, hydrogen embrittlement and loss of fracture toughness. These degradations have been more severe and widespread than could have been predicted from worldwide experience with the zirconium alloys up to the point at which they were installed in the Candu-PHW reactors. They have also occurred by unique mechanisms that have not previously been experienced. Recognizing that these unanticipated deteriorations required that pressure tubes be monitored to a degree far beyond that which is necessary for more conventional materials or called for in any existing regulations, the nuclear industry instituted intensive programs of in-service inspection, analysis of removed pressure tubes and supportive research. These programs have continued with increasing intensity to the present time.

Through the intensive work carried out since 1972, a considerable understanding of presently recognized mechanisms of deterioration has been accumulated. New mechanisms of deterioration, or variations of previously recognized mechanisms, continue to be identified and there is no doubt that advances in the knowledge of pressure tube behaviour are still to be made.

A significant finding has been the realization that the susceptibility of a pressure tube to degradation is dependent on subtle variations in material properties, fuel channel design, installation procedure, fast neutron fluence and operating environment. Consequently, the assessment of pressure tubes for "fitness-for-service" varies from reactor unit to reactor unit, and, within a reactor unit, from one tube to another. Each pressure tube must be considered individually and a case history of its initial characteristics, state of health and operating history built up over a period of time as the results of monitoring accumulate. The success to date of this individualistic approach has been demonstrated by the ability of the industry to limit the number of unexpected catastrophic failures to two, despite the severity of the deterioration mechanisms.

The analysis of "fitness-for-service" of pressure tubes has traditionally been made by the owner and his technical advisors and has been based on the state of knowledge at that time. Although a very conservative approach has always been taken, various philosophies, methods and rejection criteria have been applied. The

nuclear industry now realizes that a standardized approach to "fitness-for-service" assessment is required. Through the Candu Owners Group (COG), a major thrust is being made to elucidate this aspect of pressure tube technology.

In calling for the present study, the AECB wishes to contribute to the establishment of well defined guidelines by which the AECB (and other interested parties, outside the immediate group of technical experts currently analyzing pressure tube performance) can judge the adequacy of "fitness-for-service" analyses and can ensure that all known deteriorations and operating factors have been considered.

The term "Rejection Index" used in the title of this report is a euphemism for a "relatively simple set of criteria" which must be considered when establishing the safe limit to the operating life of a pressure tube. Since any such criteria have to be based on generic data obtained from a variety of sources, it is recognized by the authors that, at the present time, the suggested rejection index can only be used as a "flagging" mechanism. Any decision on the final rejection of a pressure tube could only be based on a detailed analysis related to that specific tube, its history and its operating environment.

In establishing a rejection index, the authors had to take careful note of present periodic inspection requirements. As a consequence of this, suggestions have been made on how the inspection requirements for pressure tubes should be revised to ensure that all proposed rejection criteria are monitored and that the results are analyzed in a consistent fashion.

This study was based on information available in the AECB library up to November 1988. A bibliography of this information is given in Appendix A. It is intended to be a critique of key issues associated with establishing a realistic "rejection index". It is recognized that some aspects may not be fully elucidated and require further work. No attempt has been made to fully reference the information, ideas and opinions expressed in this report. The authors have felt free to draw,

without reference, on a wide range of information generated by Ontario Hydro, AECL, other utilities, AECB and many other sources within the nuclear field.

1.2 Scope and Objectives

The following general objectives were set by the AECB at the outset of the project:

"For zirconium-2.5 wt% niobium used for Candu reactor pressure tubes:

survey material properties which would affect the nature and extent of pressure tube failure; and

develop an index, or a set of criteria (based on property values), which could be used for deciding whether a pressure tube should be removed from service."

To meet these objectives, the authors set themselves the following tasks:

- review information available to them on the following topics:
 - (i) Safety considerations in pressure tube failures
 - (ii) Pressure tube failure events in Canada and worldwide
 - (iii) Metallurgical degradation of zirconium alloys in reactor service
 - (iv) Operating conditions relevant to pressure tubes during normal operations, upset and emergency conditions
 - (v) Pre-service and in-service inspection practices currently applied to pressure tubes,
 - (vi) "Leak-before-break" and other safety philosophies,
 - (vii) Current requirements for inaugural and periodic inspection of pressure tubes, and
 - (viii) Current and newly developed inspection technology.
- critically review available information on pressure tube degradation mechanisms and property data to establish opinions on its validity and/or assess its weaknesses.

- identify design and operating factors which may lead to additional constraints on pressure tube life.
- review all known life limiting factors related to defect tolerance, design and operating factors and degradation of fracture properties.
- on the basis of the above considerations recommend a rejection index focused on the most important criteria.

The authors believe that, within the obvious constraints of budgeted time and information availability, they have been able to largely fulfil these objectives and tasks. In addition, issues related to establishing a rejection index have been identified, suggestions for a modified periodic inspection program have been made, and a "rejection index" has been recommended.

2.0 REJECTION CRITERIA PHILOSOPHIES

2.1 Jurisdictional Requirements

The current requirements for periodic inspection of pressure tubes are covered by CSA N285.4 Periodic Inspection of CANDU Nuclear Power Plant Components. Section 12 relates to "In-core Coolant Pressure Tubes".

When this code was first drawn up in the early 1970's, experience at NPD, Douglas Point and Pickering-A gave no indication that pressure tubes should be treated differently from other materials. Because of the small size of the rupture and its postulated consequences on a LOCA, AECL and Ontario Hydro argued against the inclusion of pressure tubes in the periodic inspection program. However, at the AECB's insistence, pressure tubes were included, but only to the degree that a supplementary inspection was carried out to audit dimensional changes that were anticipated due to neutron irradiation. These inspections were already part of a well planned Ontario Hydro program to provide generic information. They have subsequently proven very beneficial in demonstrating that in-reactor deformation has generally taken place at a much faster rate than was originally anticipated. Additional clauses related to volumetric inspection of rolled joints were added some years later, again at the AECB's insistence, to update the requirements, after a series of cracks was detected in rolled joints in Pickering Units 3 and 4. The sample size set for both these inspections (e.g., four pressure tubes in the high power region and one pressure tube in the low power region of the first reactor unit) was compatible with the limited objective of auditing generic deteriorations.

Evaluations and acceptance of indications/deficiencies are covered generally in Section 8 and for pressure tubes in Section 12.6.2 of the above code. Except for specifically designated unacceptable conditions related to rolled joint cracks, considerable freedom is given to the owner for the evaluation and interpretation of defects and deficiencies. No basic guidelines covering acceptable methods for either evaluation of defects or assessing their acceptability, are provided.

It is interesting to recall that "leak-before-break" concepts for pressure tubes were well discussed, but not credited, in setting the original inspection requirements laid out in N285.4. As demonstrated by the rolled joint leaks found in Pickering Units 3 and 4, leak-before-break permitted a powerful integrative inspection technique for all pressure tubes in post Pickering-A reactors during the first few years of service. The present authors feel that this combination of de facto leak-before-break monitoring plus the specified monitoring of a sample of pressure tubes for generic degradation(s) constituted an adequate inspection scenario for pressure tubes having the properties then anticipated. However, subsequent reactor operating experience has clearly demonstrated that the actual properties exhibited by pressure tubes are such as to require consideration well beyond that applicable to other more conventional reactor materials. This need has been recognized by the owners of Candu reactors who have put into place numerous inspections and carried out numerous detailed analyses far in excess of present requirements, but not yet recognized in N285.4.

It appears to the authors that to update N285.4 to become a credible working document on pressure tube inspection would require at least the following changes:

- (a) A statement regarding the philosophy of pressure tube inspection in terms of maintaining an acceptably low failure probability and in auditing generic deteriorations.
- (b) A statement regarding the designation of the present generation of pressure tube materials as "special" materials having more extensive periodic inspection requirements.
- (c) A recognition that due to neutron irradiation and/or deuterium pick-up, pressure tube fracture properties deteriorate during service. Monitoring of material properties is required to the same degree as defect monitoring.

- (d) A recognition that some failures and some potential failures can be avoided by correctly installed and operating fuel channel equipment. This also applies to the success of leak-before-break monitoring. Mechanical/operating inspections of key equipment must therefore be included.
- (e) A comprehensive set of guidelines as to how a wide range of presently known deteriorations and defects should be sized, evaluated and analyzed for acceptability.
- (f) A set of acceptance criteria for all known types for deteriorations.
- (g) An expanded inspection plan taking into account extra inspections when deficiencies are detected.

In considering how the above changes could be incorporated into the present N285.4 format, it appears that a general expansion of the existing paragraphs of Section 12 supported by a mandatory Appendix containing comprehensive guidelines and acceptance criteria, would be suitable. Care should be taken to define a core inspection program applicable to pressure tubes that have been correctly installed and adequately inspected during fabrication and installation. This program should be distinguished from additional inspections that may be required if a deviation in installation or operation is detected or if specific degradations are identified which need extended monitoring.

To cover all presently known degradations, the suggested mandatory Appendix would have to be a very detailed document. The Non-Mandatory Appendix-A of ASME Section XI entitled "Analysis of Flaws" provides an excellent example of the type of document required. It presents detailed guidelines under the following topics:

- Flaw model for analysis,
- Method for size determination,

- Material Properties, and
- Analysis.

Although Appendix-A, of ASME Section XI, is oriented towards steel piping and pressure vessels, the topics covered are appropriate to pressure tubes. The unique Candu-PHW technology built up for zirconium alloys would need to be substituted for the steel technology of the US reactors.

Another valuable feature of Appendix-A, of ASME Section XI is the inclusion of lower bound curves for fracture properties which are to be used when direct surveillance data is not available. Referenced are fatigue crack growth rate data and neutron irradiation effects on fracture properties of steel. This feature should be incorporated in the proposed Appendix for N285.4 and cover (at least) reference data for delayed hydride crack growth rates and the effect of neutron irradiation and deuterium pick-up/orientation on fracture toughness properties and critical crack lengths for zirconium alloys.

2.2 Alternative Inspection Philosophies

Confidence that a material will exhibit a low failure rate is dependent in part on the defect tolerance of the material and in part on the degree of inspection possible. Since pressure tube operating experience has demonstrated that defect tolerance decreases with time in service, large numbers of inspections have been necessary to provide an additional safeguard against an increase in the in-service failure rate. Up to the present time a method of performing a fully comprehensive inspection of each pressure tube is not available. Even if it were, its application to every pressure tube would represent an unacceptable economic penalty, and alternative inspection philosophies must be sought.

The leak-before-break (LBB) concept is an attractive and generally accepted basis for a global monitoring system which will be discussed later.

Probabilistic studies aimed at determining the level of inspection frequency and reliability required to ensure a suitably low failure probability have been used in aircraft inspection and in recent years have been considered for the inspection of nuclear power plants. In both these cases a single well studied slowly propagating failure mechanism is involved (aircraft = fatigue, reactor piping = stress corrosion cracking). A major difficulty with this type of analysis for aircraft is that the basic information on cracking probability, etc., applied in the analysis, is usually only being created as current inspection programs proceed. This means there is a need to update constantly the analysis to take account of the latest information. Consequently, the original predictions may prove useless if a rapidly propagating cracking mechanism is active. It also takes several years of performance feedback to establish reliable guidelines for inspection intervals.

For BWR austenitic steel piping systems, a study to determine required in-service inspection intervals for reactor piping system on the basis of probabilistic fracture mechanisms calculations concluded that this was difficult due to several factors including: inadequate knowledge of the true reliability of existing piping; the lack of established USNRC regulatory goals for safety and reliability of piping systems; and the current reliability of non-destructive examination which was often inadequate to restore confidence in an otherwise unreliable piping system.

It would be difficult to apply the probabilistic approach to Candu-PHW pressure tubes since they have multiple failure modes and mechanisms which are only just becoming understood; and for which comprehensive inspection techniques are only now becoming available. To the authors of this report, it appears that the main benefit of this approach could be gained at a later stage in setting inspection intervals and sensitivities for maximum efficiency. Efforts to apply these techniques to pressure tubes should be encouraged.

The application of a periodic overpressure (proof) test, at ambient temperature, applied simultaneously to all fuel channels in a reactor has been

considered in the past. This method takes advantage of the smaller critical crack size associated with the reduced fracture toughness at low temperatures and higher pressures. The disadvantage of the considerable maintenance penalty incurred in the event of a tube fracturing is a great deterrent to using this method. The instantaneous nature of the test is also incompatible with the requirement to detect slowly propagating delayed hydrogen cracks in pressure tubes. A variation of this test, whereby crack propagation by delayed hydride cracking is encouraged by holding a state of normal operating pressure and an intermediate temperature, has been used in several reactors to increase leak rates to an identifiable level. However, the disadvantages, as demonstrated by the inadvertent destruction of pressure tube B2N6, have been amply demonstrated. It would seem to the authors that any form of proof test should only be used as a last resort; e.g., if there was an imperative reason to demonstrate that a reactor could remain operating for a further short period of time.

2.3 Leak-Before-Break Concept (LBB)

Although LBB has been envisaged as an acceptable concept since the inception of in-service inspection programs, it has only recently been given credit in safety analyses of PWR, BWR and Candu reactor systems. Changes in regulations now allow the LBB approach to be applied in the USNRC licensing process. In its evaluation, deterministic fracture mechanics analyses and other criteria have to be met before a piping system can become an LBB candidate. Certain situations including susceptibility to failure due to stress corrosion cracking, low and high cycle fatigue, water hammer, etc. are definitely excluded. In particular, LBB analyses have been accepted for PWR primary systems with the objective of eliminating restraining devices to curb pipe-whip in the case of fracture. Ontario Hydro have benefited from these changes and have developed, and had accepted, an LBB method for application to large diameter heat transport system piping at Darlington. These recent advances indicate a general trend towards US and Canadian jurisdictional acceptance of the LBB concept, if a properly supported case can be put forward. However, in an international seminar (USNRC, Columbus, 1985, NUREG/CP-007) arranged to address international views on LBB in nuclear piping

systems, it was clear that widely different national policy positions on LBB exist. Some countries (US, Italy, Federal Republic of Germany) accept the general concept whilst most other countries either reserved judgement, or had serious reservations about whether conditions necessary to assure LBB can be expected to exist for PWR primary piping.

All participants were asked to provide answers to two questions:

- 1) What do you think leak-before-break means?
- 2) What are your critical concerns in regard to LBB?

Responses to the first question indicated the following general themes:

- "crack stability from a fracture mechanics standpoint",
- "detectable leakage", and
- "detectable leakage with the added consideration of sufficient time to respond and shut down the plant".

The following example of the wide variety of concerns expressed in answer to the second question has been chosen by the present authors as both incorporating most people's concerns and also focusing attention on critical issues such as the need to maintain a low probability of failure, even though the LBB analysis has been put forward.

"I am concerned that the leak-before-break may become a rationale for eliminating important and effective measures in the defense in depth concept (i.e., inservice inspection) along with poorly conceived measures such as pipe whip restraints. I am also concerned with confusion with LBB and low probability of failure. One should require both LBB and low probability of failure before policy changes are made. LBB provides a defense in depth measure, since the occurrence of leaks will provide a warning and opportunity to implement corrective measures to the elimination of leaks as well as potential breaks. Another concern is that LBB will lead to "overly fancy" criteria and analysis schemes, that are

based on speculation and assumptions about crack length, leak rates and leak detection. It appears that LBB requires all of the following: high toughness, low stresses, and short cracks. One must have assurance that each requirement is satisfied, and not place too much focus on any one of the requirements."

Several studies of actual pipe failure statistics indicate that a surprisingly high instance of piping breaks occur without any prior leak indication.

Rodabaugh (quoted in NUREG/CR-4469) has made estimates of the probabilities of break-before-leak (BBL) using data from both fracture mechanics models and service experience. His analysis of non-nuclear industrial piping experience suggested that the probability of break-before-leak is between 0.53 and .078. His estimate for power utility piping was given as 0.13. For intergranular stress corrosion cracking in reactor piping, about 100 incidents of break-before-leak were cited having a BBL probability of less than 0.01. For all types of reactor piping and failure modes, the BBL probability was estimated as 0.10. Rodabaugh discounted probabilistic fracture mechanics studies which gave probabilities of 4×10^{-6} for break-before-leak, as being much below that indicated by actual operating experience.

Janzen carried out an extensive study of piping failures in US nuclear power reactors for AECL. A total of 840 incidents of pipe failures (severances, leaks and defects) were considered. Whilst leak-before-break occurred in the majority of instances, the probability of break-before-leak was estimated as 0.055 for large pipes and 0.109 for small pipes. In discussing the incidence of pipe severance (or break-before-leak) Janzen pointed out that pipe failures occurring under near-normal operating conditions and caused by time-dependent phenomena such as fatigue, erosion, stress corrosion, etc. possibly in combination with fabrication/ installation faults or design errors, generally progress by finite crack growth to leakage well before failure. The nature of sudden severances precluded slow crack growth and often included forces external to the system, stress raisers or high stress regions due

to manufacturing/fabrication/installation, unknown loadings, and upset operating conditions. Instances where water hammer, over-pressure and impact had caused break-before-leak failures were given. Two instances of design error causing break-before-leak were also identified as well as a susceptibility of bellows to severance.

From the above background information, the authors conclude that even though systems may be ideal for leak-before-break detection, there is likely to be a significant number of unanticipated ruptures. The probability of these may be as high as 0.10. The addition of extra inspections during the fabrication/installation stages and/or additional constraints on upset operating conditions may significantly reduce this probability. This extra back-up, a leak detection system of proven sensitivity, a guaranteed minimum fracture toughness of the material and a proven low probability of failure are all essential features for successful leak-before-break monitoring.

2.4 Leak-Before-Break Applied to Pressure Tubes

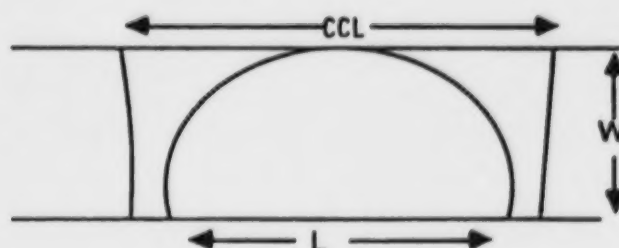
The first significant reference to LBB, as applied to pressure tubes, occurred during the design reviews for Pickering-B and Bruce-B reactors. Reference to the "defence-in-depth" provided by LBB was inserted into the design manuals at that time. This claim was based on the demonstrated practical ability to detect leaking cracks in the rolled joints of Pickering-A Units 3 and 4 which occurred after only short periods in service. No detailed study in support of these claims or to look at its (subsequently obvious) limitations was attempted at that time. In particular, the higher fracture toughness of the pressure tubes retained during the first year or two of service, and the low hydrogen levels allowing DHC at low temperature periods only, limit the usefulness of the Pickering-A experience to a demonstration of early service performance only.

In 1988, more detailed analyses of the leak-before-break capabilities of the Candu reactors were reported. The following papers provide comprehensive reviews of the current situation:

- "Leak Before Break Experience in Candu Reactors" (E.G. Price et al.),
- "Leak Before Break and Leak Detection System of Candu Fuel Channels (B.A. Shalabi et al.)", and
- "Evaluation of a Leaking Crack in an Irradiated Candu Pressure Tube (C.E. Coleman)"

The model used to show the crack dimensions at the onset of leakage, compared to the critical crack length is shown in Figure 1 below:

FIGURE 1. Leak-before-break Analysis



$$t_a = \frac{CCL - AR \cdot W}{2V} \quad \text{Equation (1)}$$

$$\text{or } CCL = t_a \cdot 2V + AR \cdot W$$

where CCL = Critical Crack Length

L = Length of Crack at Start of Leakage

AR = Aspect Ratio (L/W)

W = Tube Wall Thickness

V = Axial Crack Velocity

t_a = Time available to detect leak

The time the crack takes to propagate from the length causing the first detectable leakage to the full critical crack length is the time available for operator reaction to shut down the reactor. This is calculated from equation (1) for a symmetrical crack. In the authors' opinion a minimum time of twelve hours should

be allowed for the detection/shutdown sequence. In the previously identified reports, various cracking scenarios are described and the applicability of this type of leak-before-break analysis to DHC demonstrated in general terms. The leak detection capability of existing reactors is claimed to be adequate to meet the required time interval and even more sensitive leak detection methods have been developed for later reactors.

The authors have reviewed these reports and are impressed with the statistics of twenty-four leaks successfully detected (all originating and propagating as DHC) and only one break-before-leak (originating by a blistering mechanism and propagating as DHC). The reports make a valid case for acceptance of the general analytical approach but mislead the reader with the impression that LBB is a perfect and established methodology for all pressure tubes in all Candu reactors.

The authors feel a more balanced view would take account of limitations such as:

- certain critical information on delayed hydride cracking velocities and minimum critical crack lengths may not yet be sufficiently well defined;
- that the valid period for LBB may be limited to the first few years of reactor service;
- deficiencies in leak detection capability may exclude a significant number of fuel channels from meeting the required detection sensitivity, and
- a careful review of the situation for each operating reactor unit would be necessary to establish its status.

The critical material properties on which LBB depends are: DHC crack velocity, critical crack length and crack profile (aspect ratio L/W). Whilst establishing LBB conditions during the operating cycle is essential from the safety viewpoint, establishing LBB during shutdown would also be a major benefit in minimizing potential brittle failure.

DHC crack velocity appears to be dependent on the heating and cooling cycle, the operating temperature and the effect of neutron irradiation. These effects are shown in Figures 2-4. A prerequisite for cracking is, of course, that sufficient hydrogen/deuterium is present to form hydrides at the dwell temperature. In comparing the DHC cracking velocities with those quoted in the previously stated reports, it is noticed that the unirradiated velocity values are consistently used. If the apparent effect of irradiation is true (Figure 2) then significantly faster cracking rates should be used. Recalculation shows that in these circumstances, an LBB argument may only be sustained for short operating time periods or low fluences. This crucial point needs clarification.

For most slow failure mechanisms such as fatigue and stress corrosion cracking, the crack propagation rates increase with stress intensity. This causes acceleration of cracking as the failure proceeds. For delayed hydride cracking, however, the presently available data (Figure 3) indicates it to be independent of stress intensity. A constant crack velocity is currently used in calculating the time interval from leakage to failure. This is also a crucial issue and the authors recommend further effort be expended to establish this independence from stress intensity beyond doubt.

The acceptance of an LBB inspection scenario also entails accepting that the critical crack length (or fracture toughness) for each pressure tube in service should exceed a minimum value. This concept is not a current requirement. On the basis of the calculations provided in the previously identified reports, the CCL for a typical pressure tube inlet end at 250°C and at a hoop stress of 150 MPa, would be about 50 mm (95% lower bound limit). If faster cracking rates were substituted to account for potential neutron irradiation effects, a substantially longer critical crack length would be required for LBB. In practice, setting a minimum critical crack length (fracture toughness) would limit the allowable operating life of pressure tubes and be a major item in the rejection index.

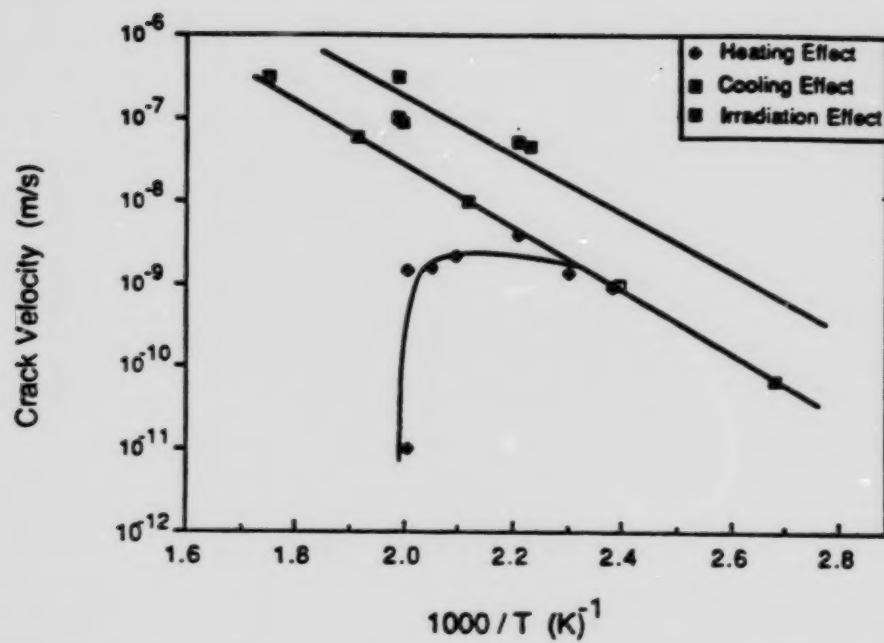


Figure 2: Temperature dependence of axial velocity showing the effect of heating, cooling and irradiation (after Coleman).

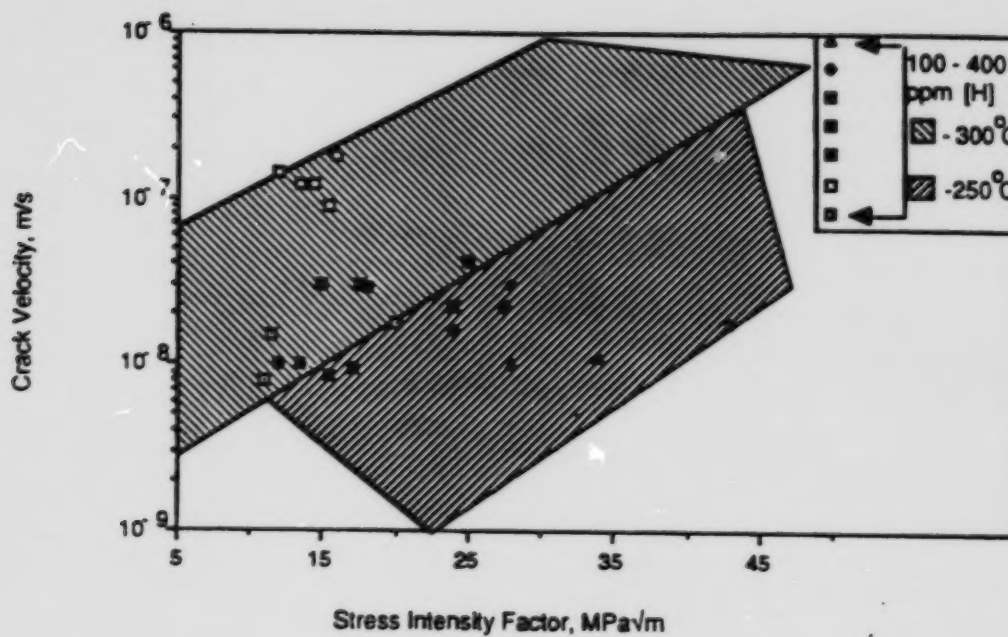


Figure 3: Axial crack velocity of Zirconium-2.5 wt% Niobium as function of stress intensity factor, temperature and hydrogen concentration (after various researchers).

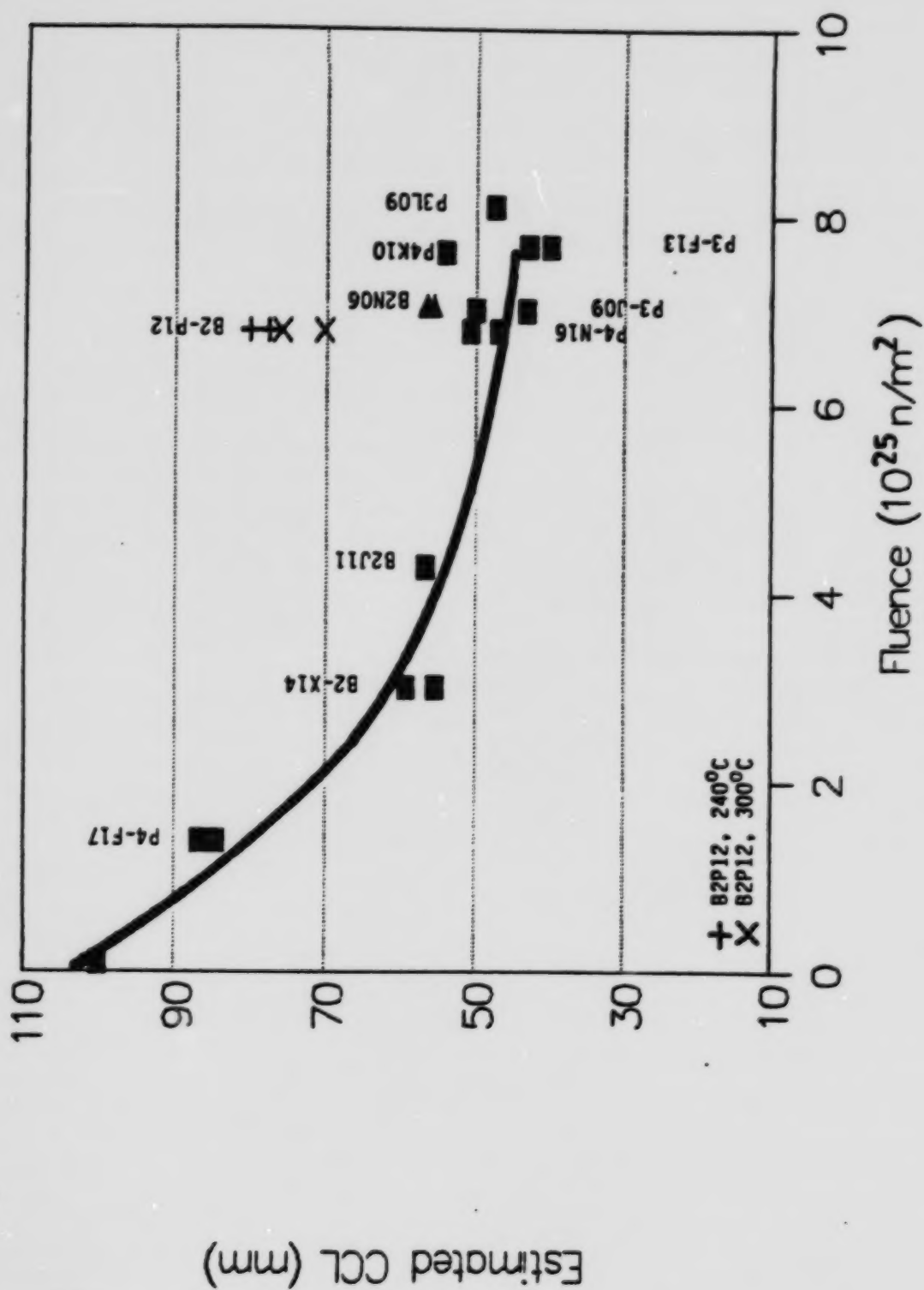


Figure 4: Decrease of CCL with Fluence.

The critical crack length (fracture toughness) of a pressure tube deteriorates in service from a new tube value of around 100 mm to less than 50 mm after several years service (Figure 4). In new tubes, hydrogen/deuterium content and the degree of hydride reorientation is known to have a significant effect on fracture toughness at room temperature and to a lesser degree at operating temperatures. The presence of moderate amounts of radial hydride may reduce the CCL to 20 mm both at room temperatures and at pressure tube inlet temperature but appears to have little effect at outlet temperatures. However, except for a very local zone adjacent to each rolled joint, significant reorientation of hydride in zirconium-niobium tubes is not anticipated since reorientation in zirconium-2½ w/o niobium alloy has been shown to occur only if the stress is significantly above the normal operating stress (E. Price, 1984).

Results from 75 slit burst tests on zirconium-niobium material showed the predominant controlling factors to be test temperature, neutron fluence, stress and crack sharpness (C.E. Coleman and C.A. Wills, 1983). The effect of circumferential hydrides was found to be undetectable. This latter finding was discussed in the report and it was concluded that the relatively few samples (two) containing both hydrides and neutron irradiation damage may have been insufficient to develop any trend within that particular study.

The relationship between CCL and fluence, as determined from pressure tubes removed from service, is shown in Figure 4. This was determined from small specimen tests which tend to give more pessimistic values than burst tests. If the lower bound value of this curve of 40 mm is valid, then it would be difficult to accept an LBB concept for pressure tubes after, in a Pickering unit for example, 10-12 years service. The determination of a completely acceptable lower bound relationship between CCL, fluence and hydrogen/deuterium effects, is essential in an LBB analysis and considerable effort should be directed toward this aim. It is not clear from the data reviewed as to the stress level used to calculate CCL. As discussed in Section 3, this stress level should match the most likely maximum stress identified in each reactor's safety report.

The authors consider that the application of the LBB model given in Figure 1 to back-calculate the minimum acceptable critical crack length, would be a valid method for determining this rejection criterion. This method has been demonstrated in relation to Zircaloy-2 tubes removed from the NPD reactor. Information relevant to each reactor unit could be substituted in the general model.

Another important consideration in the application of the LBB model is the likely profile of the cracks during growth. This has been discussed in the previously identified reports. In an idealized situation a length/depth ratio of four fits most observations, but in some instances the crack propagation may have been distorted by local stress fields at rolled joints to develop an aspect ratio as high as seven. This more pessimistic value should be assumed for calculation purposes. It is clear from trial calculations that DHC nucleated by extraneous defects such as long fabrication flaws or strings of blisters, which could potentially lead to even higher aspect ratios, are unlikely to meet LBB time limits for detection. Alternative approaches must be used to eliminate potential nucleation of long DHC cracks from other forms of defect.

The inherent leak detection sensitivity of each reactor unit varies depending on the capabilities of their particular detection system and other mechanical considerations. Improved systems designed to minimize leak detection time and to improve leak identification capability have been installed in later units. New developments are in hand to give even greater sensitivity. For complete reactor monitoring, any system must guarantee sensitivity for each fuel channel. Instances of deficiency in this regard have already been detected at Pickering-A (damaged bellows, suspected stagnation, etc.), Bruce-A (prolonged identification of leaking tubes), and Bruce-B (low flow, tight pressure tube bearings). For an LBB scenario to be acceptable, the situation for each fuel channel would need consideration and alternate inspection methods provided for deficient channels.

The authors concluded from the above background information that a valid LBB scenario can probably be demonstrated for most pressure tubes (reactor units) during early operating life . We had difficulty in establishing the operating duration for which LBB remained valid because of perceived uncertainties in critical metallurgical data. Further research effort is recommended to determine:

- an acceptable lower bound relationship between CCL, fluence and hydrogen/deuterium effects,
- the effect of flux and fluence on delayed hydride cracking velocity, and
- the influence of stress intensity on delayed hydride cracking velocity.

The authors feel that insufficient attention has been given to pressure tubes in several reactors which may be excluded from LBB monitoring because of annulus gas system monitoring deficiencies. A substitute inspection scenario for these unmonitored tubes, and for the reactor as a whole if LBB becomes invalid in later life, needs to be developed. The overall effectiveness of LBB can only be determined if a detailed LBB analysis be carried out for each operating reactor unit and it is recommended that future studies be less generic and more specific to actual reactor units.

3.0 PRESSURE TUBE FAILURE IN NORMAL OPERATION, UPSET AND ACCIDENT CONDITIONS

The term "failure" is used generally in this report and described in Section 4, has been defined by the authors in terms of metallurgical degradation and interference with reactor operation. This is a very conservative definition, since from a reactor safety point of view, only those failures which would severely challenge the reactor safety systems are of significant concern.

For licensing purposes, it is assumed that the rupture of a single pressure tube under normal operating conditions followed by the rupture of the surrounding calandria tube, can be accommodated by the reactor safety systems. To date, there have only been two pressure tube ruptures, and only one related to normal operating conditions. When pressure tube P2G16 ruptured during normal operation its calandria tube did not fail. After the annulus gas had been filled and pressurized, the pressure was relieved through the bellows assembly. Shutdown was achieved without the activation of special safety systems.

The integrity of the calandria tube is an important factor in minimizing the consequences of pressure tube rupture at full power. If it is maintained, the occurrence of a serious process failure can be prevented and damage to other core components minimized.

Burns, in his presentation to the Hare enquiry, presents an up-to-date review of this aspect of the consequences of fuel channel rupture and a series of papers by Muzumdar and co-workers at Stern Laboratories, Ontario Hydro and AECL describe the thermodynamics of pressure tube rupture and pressurization of the calandria tube as determined by experimental programs.

Burns concluded that research to date on calandria tube survivability indicates that calandria tubes are likely to survive a rupture of a pressure tube during normal

operation, as was the case in P2G16. Bruce calandria tubes have thinner walls and the margins to failure are likely to be smaller. Since there is only a small data base on the mechanical properties of irradiated calandria tube material, Burns recommended that a Bruce calandria tube be removed and tested.

Burns has also reviewed the consequences of simultaneous failure of the pressure tube and calandria tube. He concluded that pressure wave and impact effects on adjacent fuel channels will be insufficient to cause their failure. He also concluded that although some damage to shut-off-rod guide tubes may occur, there will remain sufficient undamaged shut-off rods for SDS1 shutdown.

The authors concentrated their review on a rather different aspect; that is, the effects of pressure and temperature cycles anticipated during the lifetime operation of pressure tubes and their possible influence on pressure tube failure. Conditions pertinent to Darlington NGS were used as being typical.

The most probable events which can occur are the normal and upset conditions (levels A and B, ASME III Code) but the most serious challenges to pressure tube integrity are likely to come from emergency and faulted conditions (levels C and D, ASME III Code).

The authors reviewed the Darlington Design Manual and other readily available information to see how the designs had taken account of these transients in their consideration of pressure tube fracture properties (critical crack lengths) and leak-before-break philosophies. Whilst both these latter items are considered in the Design Manual, only a typical operating situation under nominal operating pressure and temperature are described. This poses serious questions related to design philosophy. Should the performance of a flawed pressure tube be analyzed under the worst anticipated operating conditions? Is it necessary to demonstrate the pressure tube has a sufficiently long critical crack length to permit leak-before-break under the most pessimistic combination of high pressure and low temperature predicted?

These questions need resolution. The authors feel that the answer should be yes as far as normal and upset conditions are concerned since major transients can be expected to occur many times per year. There is a possible case for leaving out emergency and faulted conditions since these are predicted to occur less than once per year.

A quick check on fracture analyses and predictions of critical crack length in generic studies, suggested that a nominal pressure (stress) is also generally used. The authors feel that for this reason they may give optimistic predictions of actual performance.

The authors recommend that these concerns be addressed and a philosophy regarding operating transients be established.

4.0 PRESSURE TUBE PERFORMANCE

The main source of information used in reviewing worldwide performance of pressure tubes is an excellent and recently published report issued by the UKAEA in which R. J. Haslam discusses available pressure tube statistics for reactors worldwide (including Canada). Haslam describes the main design features of each reactor type, summarizes operating experience and describes the circumstances and diagnosis of each major failure. Additional information related to the RBMK reactors in Russia and PRTR in the US has also been included. A much wider information base relating to Canadian reactor experience and the history of pressure tubes in each Candu PHW reactor has been reviewed from numerous AECL and Ontario Hydro reports.

For the purpose of this study the authors have chosen to categorize the performance of pressure tubes in the following manner:

- Category A: Degradations involving fracture or leakage of pressure tubes in reactors during operation. All incidents in this category could be designated "failures".
- Category B: Degradations involving fracture or leakage of pressure tubes in reactors in the shutdown state and/or defects/deteriorations detected by in-service inspection that required pressure tube replacement or other rectification before the reactor was allowed to return to service. Some incidents in this category can be designated as "failures", others as "aborted failures".
- Category C: Degradations involving defects, deteriorations, reduced material properties, design deficiencies, incorrect installations, operating deficiencies or other malfunctions (exceeding criteria such as those described in this report) which may have prejudiced the useful life

or significantly increased the risk of failure of pressure tubes, but which have not yet led to a decision on pressure tube replacement. All incidents allocated to this category may be designated "potential failures".

Category A, B and C failures are given in Appendix B. Category A failures have fortunately been very rare. Because of their consequences they have received a lot of attention and all circumstances regarding their occurrence have been analyzed in great detail. Category B failures have been quite prevalent, especially after a significant number of years of operation. Many degradations in this category have been discovered when expanded inspection programs have been applied after a Category A failure has been experienced. Category C degradations have, up to the present time, received little emphasis since from a Candu operator's viewpoint they are not yet really "failures" and are outside regulatory requirements published to the present time. The authors wish to draw attention to the very high proportion of all pressure tubes that fall into this category. It is to this category that the present study regarding a Rejection Index is directed, since it is clear that the probability of Category A and B incidents is greatly increased if the allowable limits for Category C deficiencies/degradations are incorrectly set.

In reviewing the background information on failures, the authors came to the following conclusions:

Category A (See Appendix B for details): These were, without exception, related to previously unsuspected mechanisms of failure. Error played a substantial role in their occurrence (e.g. operating error: Lucens, Chernobyl-4; design error: Pickering Unit 2, PRTR; installation/fabrication error: Pickering Units 3 and 4, RBMK's, Bruce 2). Prior indication of failure from inspection was not possible, because of the transient nature of failure in the case of operating error and because adequate inspection programs or techniques were not in place for monitoring the unsuspected defects such as rolled joint cracks, blistering or tube fabrication defects.

Undoubtedly, the probability of incidents due to all mechanisms, except operating error, could have been reduced by the application of a better quality control program during fabrication/construction and some form of auditing inspection during operation.

Category B (See Appendix B for details): The majority of these incidents were detected as a result of monitoring programs for suspected deterioration mechanisms. Left undetected, they would likely have resulted in an increase in Category A failures. The rapid development and application of suitable monitoring programs and inspection techniques has played an important part in minimizing Category A failures in Candu reactors.

Category C (See Appendix B for details): Again, the detection of these deteriorations has largely been the result of planned inspection programs. Deficiencies detected in both material properties and degradations are prerequisites for more substantial failures. A high proportion of tubes in the existing Candu reactors exhibit one or more types of deterioration to a varying degree of severity.

5.0 POTENTIAL LIFE LIMITING DEGRADATIONS

5.1 General Considerations

The authors have reviewed a considerable volume of information available on recognized degradation mechanisms and have attempted to establish opinions on its validity and/or assess its weaknesses. Special note has been taken of recent submissions to the Hare enquiry, a Fuel Channel Technology Seminar held in 1985, a review of information on the degradation of zirconium alloys available from non-Canadian literature and various overviews related to confidence in Candu fuel channels issued by AECL, Ontario Hydro and COG. Each degradation mechanism has been treated separately in the following sections, although it is recognized that considerable interrelationship exists between these arbitrarily chosen divisions.

5.2 Corrosion, Oxidation and Deuterium Pick-up

Corrosion/oxidation of zirconium alloys in heat transport system environments results in both metal loss and deuterium (D_2) pick-up. Information on this subject relevant to Candu conditions has been extensively reviewed recently.

Zircaloy-2 pressure tubes removed from Pickering Units 1 and 2 and NPD have been shown to exhibit an accelerating rate of corrosion (breakaway) and deuterium pick-up when exposed for periods of service exceeding 2000 effective full power days operation. Due to temperature and neutron flux profiles along each pressure tube, maximum oxidation and deuterium pick-up occurs towards the outlet end of the pressure tube. At the time Zircaloy-2 pressure tubes in Pickering Units 1 and 2 and NPD were withdrawn from service maximum oxide thicknesses of about 70 μm had been reached together with deuterium levels of up to 250 ppm (equivalent hydrogen). Although the increase in oxide thickness during service represents a significant metal loss, any loss in strength due to this relatively uniform wall loss is of little practical concern, having been more than anticipated by the

original corrosion allowance combined with a substantial increase in tensile strength (20-40%) due to neutron irradiation hardening. However, deuterium pick-up to the levels reached, is of considerable concern since this has substantially contributed to the failure mechanism.

For zirconium-niobium pressure tubes, experience has shown both the oxidation rate and the deuterium pick-up rate to have, up to the present time, remained low compared to Zircaloy-2. However, recent samples taken from tubes in Pickering (e.g. P3L09) indicate that some tubes develop deuterium levels well above the average. There is concern that this might indicate the onset of breakaway hydriding for zirconium-niobium alloy, a situation not previously experienced and which is expected to result in an increased deuterium pick-up rate. Alternative pick-up routes for these tubes are also being investigated and until these are ruled out, it is premature to assume that the high D_2 level is due to the onset of breakaway oxidation. Pick-up directly from the annulus gas appears to be a possible mechanism.

Although several models have been put forward for predicting oxidation rates and deuterium pick-up rates, considerable doubt exists regarding the long term performance of zirconium-niobium and in particular the possible onset of breakaway hydriding. A relationship between D_2 pick-up and oxidation rate and the levels of minor impurities such as iron in pressure tubes has been postulated and if proven could be used as a basis for selecting pressure tubes that may exhibit the highest pick-up rates for future sampling.

Since it is known that oxidation in some pressure tube alloys is sensitive to the presence of dissolved oxygen or other radiolytic oxidizing species, it has been suggested that the oxidation rate for boiling channels in the Bruce and Candu-600 reactors may be significantly different to the non-boiling Pickering units, since boiling acts to remove dissolved hydrogen from the water. Boiling could also accelerate the onset of breakaway hydriding. Information on the effects of boiling is scarce and

efforts are in hand to increase the number of samples of pressure tubes from boiling reactors available for chemical analysis.

An interesting contradiction to the above current Candu thinking, is the experience with Zircaloy-2 tubes in SGHWR reactor in the UK. By 1985 this reactor had operated in the boiling mode for over 3000 effective full power days. A pressure tube removed in 1985 showed oxide thicknesses of 40 μm to 90 μm on the inside of the tubes. However, hydrogen levels had remained relatively low (30-60 ppm) when compared to Zircaloy-2 experience in Pickering 1 and 2 and NPD, where very large deuterium pick-ups occurred in non-boiling conditions. This suggests that details of the heat transport fluid chemistry may also be extremely important.

Two other routes for deuterium pick-up are known. Diffusion of deuterium through the end fitting has been recognized as a major contributor to the build-up of deuterium in the region of rolled joints. A model for predicting this build-up has been checked against levels found in tubes removed from various reactors and found to be valid. Levels as high as 750 ppm (equivalent hydrogen) are predicted to occur over 30 years service. The effects of these high levels is discussed in Section 5.4.

Investigation of the circumstances of the Pickering Unit 2 G16 failure suggested that direct pick-up of deuterium from the annulus gas may have occurred along the length of the pressure tube. It was observed that the quantity of deuterium found in parts of the pressure tube exceeded that which could have been generated by the amount of aqueous oxidation that occurred with the heat transport heavy water. The factors involved with this mechanism and the likely rates of pick-up are being currently investigated but rates are likely to be lower than that due to oxidation.

The authors concluded from the above that an auditing program designed to regularly monitor the rate and/or level of deuterium build-up along a number of pressure tubes and their rolled joints by scrape sampling, must be an essential

feature of a future inspection program. An initial frequency of not greater than three years is thought prudent, since at the current time there are considerable uncertainties about the rate of build-up and the onset of breakaway conditions. Annual monitoring with an expanded sample base is suggested as appropriate in the period subsequent to breakaway if this, in fact, occurs. The selection of tubes for sampling should take into account compositional differences between tubes and the sample size should be sufficiently large to give confidence that representative levels have been monitored.

There are several possible deuterium/hydrogen levels that could be specified as the basis of rejection, depending on specific circumstances. The implication of each one of these will be discussed in subsequent sections. A selection is given in Table 1. A general rejection limit of 100 ppm is recommended for pressure tubes in defect free channels installed with low stress rolling procedures and with correctly positioned garter springs. The arbitrary level was chosen as a conservative maximum level after which further service could only be justified on the basis of detailed review. In some circumstances, as described in Table I and Sections 5.4 to 5.12, lower limits may have to be set to compensate for installation or other deficiencies.

TABLE 1
Possible Rejection Limits Due to Hydrogen/Deuterium Pick-Up

Possible rejection limits for D_2/H_2 (hydrogen equivalent)	Significance	Possible Application as Rejection Limit
18-24 ppm	Threshold for blistering.	In the event that garter springs are displaced and known pressure tube/calandria tube contact exists.
~30 ppm	Threshold for delayed hydride cracking at pressure tube inlet end	High Stress over-rolled joints made in error, especially applicable in early service or presence of known defects which might initiate DHC.
~60 ppm	Threshold for delayed hydride cracking at pressure tube outlet end	High stress over-rolled joints made in error, especially applicable in early service or presence of known defects which might initiate DHC.
>60 ppm	Level above which radial hydride significantly effects fracture properties at room temperature and inlet end operating temperatures	High stress over-rolled joints made in error leading to zones of reorientated hydride at pressure tube ends.
20-80 ppm	Maximum levels currently found in zirconium-niobium tubes and rolled joints.	Current reactor experience.
100-200 ppm	Level above which circumferential hydride is likely to reduce ambient temperature fracture toughness	Maximum conservative value for correctly installed pressure tubes.
200-300 ppm	Levels reached in zircaloy-2 tubes in Pickering Units 1/2 and WPD which in combination with irradiation have led to low critical crack lengths	Insufficiently conservative to use as rejection limit. Arbitrary value of 100 ppm suggested.
760 ppm	Satisfactory hot pull out strengths of rolled joints containing high hydrogen levels	Insufficiently conservative to use as rejection limit. Arbitrary value of 200 ppm suggested.

5.3 Fabrication Defects

Although a relatively high standard for fabrication inspection has always been applied to pressure tube manufacture, a number of long lap-type defects have now been found in tubes installed in reactors. Investigation has identified that pressure tubes fabricated from material originating near the top of the ingot are more likely to be susceptible to this defect. A program of retroactive reactor inspection concentrating on top end material has been initiated to eliminate as many such defects as possible.

Experience has shown that the traditional methods of ultrasonic inspections applied during the fabrication process are insufficiently sensitive to these defects because of both their orientation at a low angle to the tube surface and of their particular nature. It has also been demonstrated that in new material many of the laps are filaments of material of a different metallurgical structure rather than physical defects. It is only after exposure during service that corrosion and thermal cycling causes the lap to open up and act as a good reflector to ultrasonic inspection. It is only very recently that a new high frequency technique, sufficiently sensitive to this type of defect, has been introduced.

As described in Appendix B, a number of these lap-type defects have initiated delayed hydride cracking during service. The severe failure of B2N06 was initiated by one of these defects. There is a significant possibility that very long DHC cracks could initiate at this type of flaw, making any tubes containing them much less likely to display LBB characteristics.

The authors concluded that due to the difficulties in identification, a number of tubes with this type of defect will remain undetected in existing reactors even if retroactive inspections are carried out. These may initiate BBL failures. Defects of this type are completely unacceptable, whenever found. The application of the latest

inspection technology to the newly fabricated tubes will help to eliminate the problem for future reactors.

5.4 Rolled Joint Pull-Out Strength

Predictions indicate that the levels of deuterium in rolled joints will increase with service life reaching (up to) 750 ppm after 30 years. A study of pull out strengths of rolled joints, deliberately hydrided to values well above 750 ppm indicate that the rolled joint always retains adequate strength. Models available to calculate this build-up reasonably predict current service experience. The authors conclude from the information available that no special non-destructive testing of rolled joints is required for this purpose. Periodic monitoring of equivalent hydrogen levels of selected rolled joints will give adequate warning of excessively rapid build-up and of unacceptable levels of deuterium. Regions of the pressure tube just inboard of the rolled joint which are more likely to lead to a loss of strength in the event of excessive hydriding should be specially monitored. In the absence of any guidelines in other literature, an arbitrary limit of 200 ppm equivalent hydrogen is recommended by the authors as a conservative maximum level that can be tolerated without concern about reduced pull out strength. This limit could possibly be increased in the future in the light of more experimental information.

5.5 Fracture Toughness

The background to the fracture toughness of pressure tubes has been discussed earlier (Section 2.4). At the present time there is no jurisdictional requirement for pressure tube material to meet any fracture toughness level. This appears to the authors to be out of step with general (international) practice for nuclear materials, particularly for those materials to be exposed to neutron damage and for which deterioration of fracture properties is to be anticipated.

There are several possible approaches to setting a minimum fracture

toughness limit. However, it is recommended by the authors that, initially, an LBB scenario be a requirement for pressure tubes and that minimum fracture toughness requirements be set with this in mind. Consequently, the basis of setting the limit should be the minimum critical crack length for any pressure tube in a particular reactor, that is required to fulfil the LBB detection scenario described in Section 2.4. Specific leak detection sensitivities and other operating conditions related to the particular reactor unit should be applied in this calculation.

The authors recommend that a "minimum critical crack length" rather than a minimum "fracture toughness" be the requirement since this is both directly related to the LBB model and takes into account the variables of stress, fracture toughness and temperature, applicable to different locations in the pressure tube and rolled joints.

A practical difficulty exists in measuring the actual fracture toughness of any particular pressure tube, since this involves destructive testing. A relatively small data base of tubes exposed for a significant number of years service exists at present, but a program of periodically removing sample tubes from a number of reactors is in hand which should considerably enlarge this data base in the next few years.

It is recommended by the authors that an interim reference set of fracture toughness/critical crack length properties be established, as representing lower bound values. All pressure tubes should be judged against these curves, on the basis of their known neutron fluence, operating conditions and the results of scrape samples (D_2 analysis) taken directly from them. In the event that the calculated critical crack lengths approach the LBB limit, actual fracture properties should be measured on sample pressure tubes removed from the reactor for that purpose.

The above approach is deficient in being unable to recognize any deleterious effects from radial hydrides. However, these are only likely to be present in limited regions adjacent to the rolled joints and their effect will be small until deuterium

levels exceed 60 ppm.

Considerable research is currently being carried out to rationalize fracture toughness values measured by different fracture test methods. It is likely to take a few more years before a precise correlation is determined. In the meantime, it is recommended that the most conservative value, whatever the defensible test method, be assumed to hold.

Another important requirement (besides the LBB requirement) is that a pressure tube retain sufficient fracture toughness to tolerate many other types of defects other than DHC which may be present or develop during service. In practice, the acceptance of the minimum critical crack length limit described above provides an adequate tolerance to most other defects occurring in-service.

5.6 Delayed Hydride Cracking

This subject has been extensively researched since the Pickering rolled joint cracks occurred in 1975. The prerequisites of DHC cracking are well established as: a stress equal to or exceeding the yield stress at a smooth surface, or a stress intensity exceeding $4.5 \text{ MPa (m)}^{1/2}$ at a stress raiser; the presence of sufficient hydrogen and deuterium to exceed the terminal solid solubility at the cracking temperature; and a temperature cycle to allow hydrogen redistribution under the stress field. Cracking is preceded by an incubation period which increases with lowering stress intensity and temperature. The velocity of delayed hydride cracking is discussed in Section 2.4. It is extremely slow at room temperature but significant at temperatures above 60°C . Models exist which reasonably predict the main features exhibited by test specimens and pressure tubes. The main concerns that remain are identified in Section 2.4 as: uncertainty as to upper bound velocity values to be assumed for LBB calculations; and a need to confirm the independence of velocity from stress intensity.

The elimination of high residual stresses at rolled joints in all reactors constructed since 1975 has apparently been successful in eliminating the original cracking problem. Cracks in Bruce Unit 2 rolled joints, for which stress relieving

was used to reduce stresses in over-rolled joints, have been diagnosed to have initiated in the period before the stress relieving was carried out. Any significant reoccurrence of delayed hydride cracking in pressure tubes is most likely to be as a secondary cracking mechanism, initiated by other types of defect, whose dimensions provide sufficient stress intensity under the operating stresses. Examples of DHC initiated by quite severe fabrication defects are reported in Appendix B. It is possible that as deuterium levels exceed the terminal solid solubility (TSS) at operating temperatures, this type of secondary cracking may occur more frequently in the future. For this reason, crack-like defects of any size present in a pressure tube must be considered unacceptable. At some point in time (depending on deuterium pick-up rate and stress intensity) sharp defects will inevitably initiate DHC failure and because of their significant length, negate the LBB scenario.

The authors carefully considered the possibility of excluding very small cracks from this limit. However, they concluded that because of the small size of the cracks which could initiate DHC under certain conditions; the practical difficulties in accurately sizing small defects; and the inadequacy of present information on the effect of oxide wedging in increasing the stress intensity at the crack tip; this would not be prudent at the present time.

The likelihood of the minor service defects of the type currently being found in pressure tubes, actually initiating failure, may be exaggerated if a simple stress intensity/applied stress relationship is applied. Figure 5 shows the relationship between applied stress and crack depth for semi-elliptical cracks. Materials of various K_{1H} values are considered. It would be anticipated from this graph that fret marks, fuelling score marks and other mechanical damage known to be present on the surface of pressure tubes in operating reactors would have initiated failures. This, in fact, has not occurred. Experiments to try to deliberately initiate cracks at surface score marks suggest that this is unlikely at operating stresses. Other laboratory experiments on shallow surface defects have also demonstrated that they have a low probability of nucleating DHC at operating stresses. Fuel fret marks, commonly found in all operating pressure tubes, are significantly deeper than score marks but being of quite smooth profile generate a lower stress intensity. Fret

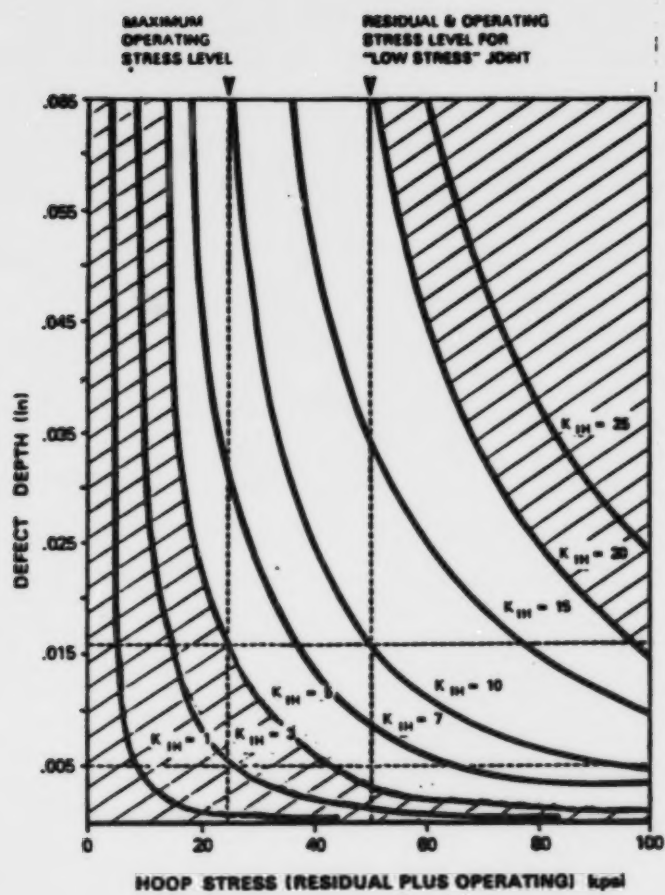


Figure 5 Critical Flaw Sizes for Zr-2 1/2 Nb of Various K_{IH} Values

marks, up to the present time, have not initiated DHC. Different patterns of radial hydrides have been found to precipitate around defects loaded to the same nominal stress intensity but having different aspect ratios. Long shallow defects showed hydrides forming a radial pattern within the elastic plastic boundary at the root of the defect. For short deep defects, truly radial hydrides were formed. Only the latter initiated DHC. Further experimentation is required to determine the dimensions of various types of defects required to initiate DHC. Acceptable limits of length and depth for score marks, fret marks and other small defects other than cracks can then be established.

The authors feel that a limit to the acceptable depth of fret marks, score marks and other surface imperfections is required, but do not have available the necessary information to suggest limiting values. It is recommended that more analysis and experimental studies be done on this topic.

One possible way to eliminate DHC completely as a potential failure mechanism occurring during reactor operation is to limit deuterium/hydrogen content along the pressure tube to below the hot solubility limits. These values are likely to be exceeded after only a few years service and if applied as a limit could significantly shorten allowable pressure tube life. Consequently, the authors recommend an alternate approach based on: establishing an LBB scenario; sensitive fabrication inspection to ensure the absence of lap type defects, installation inspection to indirectly demonstrate rolled joints of low residual stress; and an in-service inspection of a sample of tubes to audit the dimensions of typical surface degradations that may occur during service. This combination should provide adequate safeguard against unexpected DHC failure whilst allowing hydrogen/deuterium to rise to the levels suggested in section 5.2.

5.7 Blister Formation

Since the sudden failure of pressure tube P2G16 during service in 1983, a considerable amount of effort has been focused on understanding the blistering mechanism. Pre-requisites for blister formation are: sufficient deuterium/ hydrogen to exceed the solid solubility at the minimum temperature of the blister region and a steep temperature gradient induced by locally cooling the surface. Modelling of heat transfer occurring with various scenarios of pressure tube/ calandria tube contact, has been successful in predicting the characteristics of blistering observed in service. Contact temperatures of 203 to 236°C are predicted with threshold hydrogen levels of 18 to 24 ppm. Extension of the models to predict the probability of blister formation in other reactors has been carried out for different assumed garter spring displacements, deuterium pick-up rates, initial hydrogen contents, etc. These predictions suggest blistering will take around nine years to occur in pressure tubes in Pickering 5/6, Bruce 3/4 and Pt. Lepreau GS and that 4 to 20 blistered tubes will have developed after twelve years service. These predictions are very dependent on assumptions about deuterium pick-up rates. If breakaway hydriding occurs for pressure tubes (this is not determined at the present time) the rate of blistering may accelerate at some future time.

The key ingredient for blister formation is pressure tube/calandria tube contact caused by a design flaw in the garter springs which allowed their displacement during reactor construction and commissioning. This has now been corrected in the most recently tubed reactors by the use of a redesigned "tight" spring and by repositioning displaced garter springs in five reactors before they were placed in service. Some twelve units are still susceptible to the blistering problem.

The sudden failure experienced with P2G16 has been interpreted as developing in three stages. The first stage involved the initiation and growth of blisters to a critical size, the second stage involved the propagation of delayed hydride cracking at operating temperature to form a low aspect-ratio, non-leaking

critical crack and the third stage involved fast brittle rupture of the tube. The high likelihood of multiple blister formation, low-aspect cracks and rapid delayed hydride cracking rates at operating temperature all suggest that LBB is unlikely to apply to tubes failing by this mechanism. Consequently, if pressure tube contact cannot be avoided (i.e. displaced garter springs), limiting acceptable deuterium/hydrogen levels to below the appropriate threshold value (see above) is recommended by the authors as a control measure to prevent these failures. Periodic inspection of a sample of pressure tubes to check for the development of blisters is also recommended.

Current inspection techniques are only sensitive to "cracked" blisters but not "uncracked" blisters. It is also difficult, at the present time, to size blisters in pressure tubes, in order to benefit from the interval between initiation and critical size for DHC initiation which may be several years in duration. More work is required to define: limiting blister sizes, the risk factor involved in leaving any known blisters in service and if bulk deuterium levels above the threshold level are acceptable in some circumstances, because of low blister growth rates.

In reviewing this information, the authors concluded that: failures initiated by blisters are unlikely to be detected by "leak before break" and that for reactors with displaced garter springs, both scrape sampling for deuterium level and sample volumetric inspections to detect blister formation are essential. Recommended rejection limits for blistering are: displaced garter springs; and the presence of any blister. At some later stage, it would seem likely that a maximum blister size based on diameter, depth and proximity of blisters may become acceptable.

5.8 Irradiation Induced Elongation

Cumulative data obtained from seventeen years of reactor operation have shown a linear relationship between pressure tube elongation and neutron fluence. Models have been developed to explain this mechanism in terms of crystallographic features and preferred orientation due to fabrication. The authors concluded that

since the elongation rates expected are well established, each reactor unit can be monitored by dimensional measurement of a small sample of tubes. As allowable mechanism limits are approached, an expanded sampling would be required to ensure tubes exhibiting maximum rates were included in the program.

The principal limit for elongation is allowable bearing travel. To maximize this travel, tubes are allowed to elongate at one direction and then the other. It is recommended that an initial limit be set at 75% of the available bearing travel in each direction. Limits as high as 90% could be allowed if each individual channel is measured. A secondary limit needs to be applied to feeder spacing. This spacing must be maintained to prevent feeder fretting. Differential elongation between adjacent fuel channels could conceivably reduce the feeder spacing and a minimum gap of 6 mm is recommended.

5.9 Irradiation Accelerated Diametral Creep

Well established models relate diametral creep to temperature, neutron flux and time and adequately predict the range of diametral creep rates experienced over the last fifteen years. The authors concluded that because of these models, each reactor can be monitored for diametral creep by periodic dimensional measurement of a limited sample of tubes.

Limits to diametral creep must be established to ensure:

- that the diametral creep does not exceed the metallurgical rupture ductility of the pressure tube material
- that the diametral expansion does not significantly alter the pattern of coolant flow over the fuel and:
- that the initial clearances between the garter spring, pressure tube and calandria tube are not lost (garter spring lock-up) leading to surface damage.

A recent study has established that the likely rupture ductility for zirconium-niobium pressure tubes is greater than 5% diametral expansion. This value is likely to be conservative with respect to the other two criteria for most reactors and is therefore recommended as the diametral creep limit. An exception to this may be Bruce A Unit 3 where garter spring arrangements may lead to the latter limit being exceeded.

Wall thinning due to pressure tube elongation and diametral creep is estimated at about 6% over the reactor lifetime. It is not considered as a limit since this reduction plus corrosion should be more than compensated for by the irradiation hardening of the pressure tube during its early service.

5.10 Irradiation Accelerated Sag

Models existing for pressure tube sag accurately predict fuel channel deflections obtained from in-service measurements. However, there may be further work required to improve the models for use in predicting pressure tube/ calandria tube contact. The authors conclude that because of the adequate models, each reactor can be monitored for sag by periodic measurement of a limited number of tubes.

Limits to acceptable sag must ensure:

- contact does not occur between the pressure tube and the calandria tube.
- sagging calandria tubes do not come into contact with other horizontal reactor components and:

The former limit represents a physical deflection of perhaps 50 mm, progressing over a predicted period of more than ten years. The latter limit has significance in the first years of operation if garter springs have been displaced from their intended position and in a few reactors after many years service due to normal pressure tube sag.

5.11 Fatigue Cracking

There have been no instances of fatigue cracking experienced in pressure tubes and because of this, fatigue has received relatively little research attention. A recent review of fatigue in zirconium alloys consolidated available information. Little specific data relates to zirconium-2½ W niobium pressure tubes. Data on

O

endurance limit curves and on crack growth rates in air has been reported. Recent tests carried out on irradiated material show that hydriding has little effect on crack growth rate. Important factors such as the effect of water environment have not yet been fully studied.

Recent interest in fatigue has centred around the possibility of fatigue damage occurring during SLAR adjustment of garter springs. This is thought unlikely to occur if surface defects present at the pressure tubes surfaces are no worse than those generally present. Acceptable flaw geometries (in terms of length and root radius) for both SLAR operation and normal operation have been determined.

On the basis of the above information, the authors conclude fatigue cracking is unlikely to initiate from the usual range of shallow surface flaws existing in pressure tubes. Further work is still required to determine fatigue properties and to define acceptable length/depth dimensions for typical surface defects. If fatigue cracks do initiate and propagate through the wall they are likely to cause failure in an LBB manner. A mechanism by which fatigue cracks initiate delayed hydride cracking has been demonstrated in laboratory tests and can be envisaged for operating pressure tubes.

5.12 Construction and Operating Deficiencies

Both major mechanisms of pressure tube failure; rolled joint cracking and blistering, can be attributed to deficiencies introduced during design and/or

construction. Incorrect operation and/or poor flow conditions in annulus gas systems are suspected of increasing deuterium pick-up and preventing sufficiently sensitive leak detection capabilities. Optimum settings of bearing travel, feeder clearances and other mechanical equipment is required if pressure tube distortions are to be accommodated without creating other problems.

The authors recommend that periodic auditing of these critical system parameters be recognized as an important requirement and be included in the periodic inspection program. Deficiencies detected should require detailed investigation and it is also recommended that additional inspections for blistering, deuterium build-up, dimensional interference, fracture property degradation, etc., caused by these deficiencies, be implemented.

6.0 RECOMMENDED BASIS FOR REJECTION INDEX

On the basis of this background information, the authors have selected the following criteria as representing the most important factors in establishing a Rejection Index.

6.1 Limits Related to Design Intent and Normal Operation:

- All garter springs must be in place, remain within spacing tolerance and be designed to prevent pressure tube/calandria tube contact;
- Annulus gas system must be demonstrated as currently circulating, being purged to prevent accumulation of D_2 and sensitive to potential leaks in all fuel channels; and:
- Pressure tube dimensions and distortions must be limited to maintain the fuel channels within the original design intent.

6.2 Limits Related to Defect Tolerance:

- Adequate leak before break sensitivity in terms of detection limits, response time and action time must be demonstrated for all fuel channels;
- Long lap type fabrication flaws are unacceptable;
- Crack-like defects of any size are unacceptable; and:
- Score marks, fret marks and other defects with contoured profiles must fall below certain depth, length and stress intensity limits to be determined by further study.

6.3 Limits Related to Property Degradation:

- Each pressure tube must be demonstrated to have a critical crack length at operating temperature sufficient to ensure leak before break in most circumstances;
- The maximum deuterium/hydrogen level in any pressure tube should not exceed a certain level between 20 and 100 ppm which will be set according

to the known history of that tube;

- The maximum deuterium/hydrogen level in any rolled joint should not exceed a limit which is currently recommended as 200 ppm; and:
- The maximum diametral creep strain should not exceed a limit which is currently recommended in the design and safety reports.

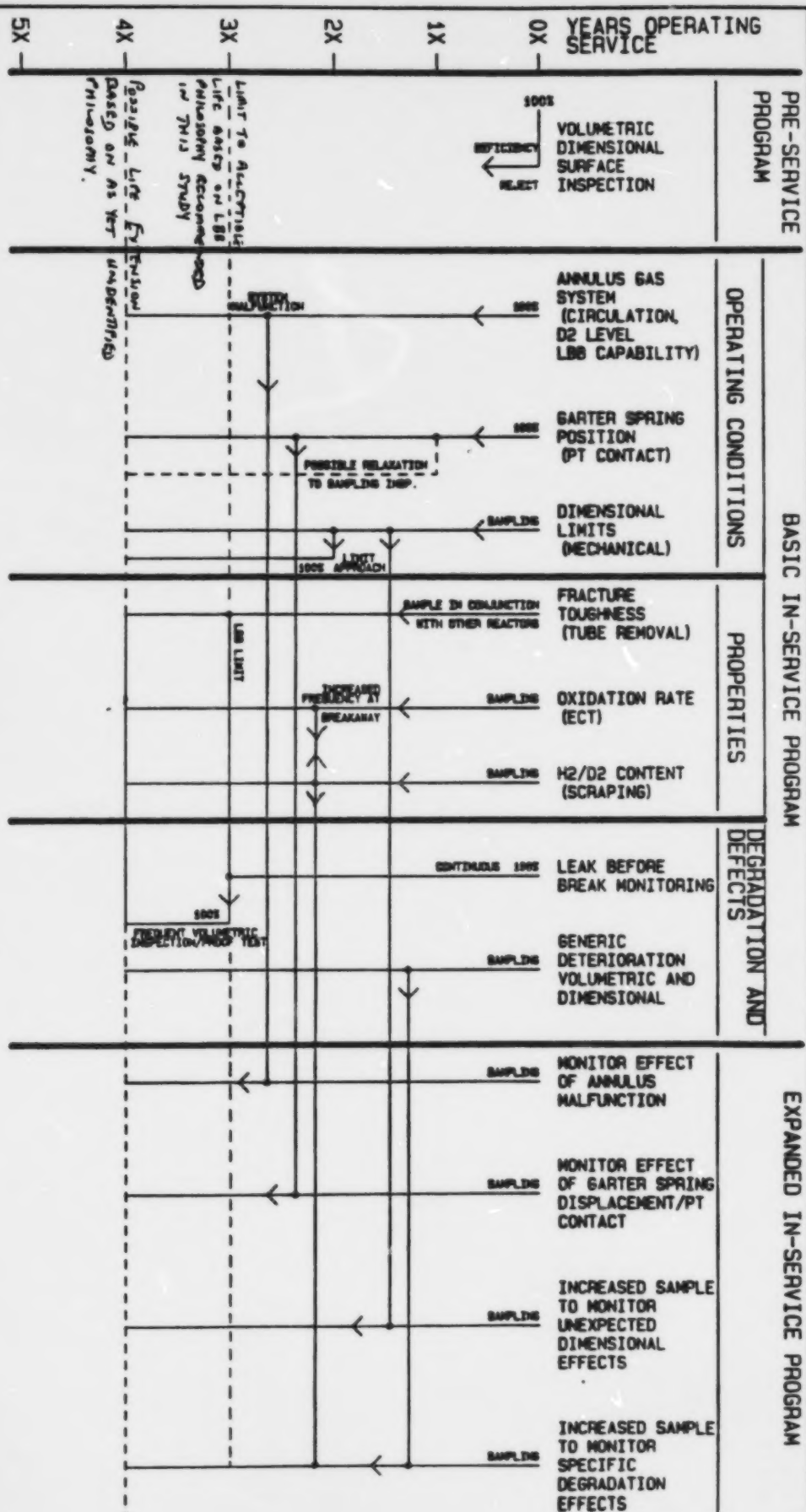


Figure 6: SCHEMATIC PRESSURE TUBE INSPECTION PROGRAM

7.0 SCHEMATIC INSPECTION PROGRAMS

In recommending appropriate limits for the Rejection Index, it has been necessary also to consider carefully, possible inspection scenarios. These inspections, selected from the previous sections, have been assembled into a schematic pressure tube inspection program (Figure 6). The authors suggest that this may be useful in establishing a future program for reactor pressure tubes.

The total program perceived includes a "pre-service" program, a "basic" or "core in-service" inspection program (designed on the basis that leak-before-break can be established as the major monitoring technique) and an "expanded in-service" program to include additional monitoring when defects or malfunctions are detected.

8.0 GENERAL CONCLUSIONS

- 8.1 Considerations of the in-service performance of pressure tube materials indicate that the probability of failure is considerably higher than for other Candu reactor materials.
- 8.2 The degree of inspection and the level of rejection applied must compensate for this increased unreliability.
- 8.3 Economic and operating restraints make global inspection of pressure tubes with external inspection devices unattractive.
- 8.4 Pressure tube leakage ("Leak-before-break") appears to be the most attractive indicator of pressure tube degradation, if it can be demonstrated as valid.
- 8.5 Although "leak-before-break" capability has been demonstrated in general terms for pressure tube leak detection, a number of uncertainties remain regarding its application to Candu reactors. These relate to vital property data, crack initiation from other types of defect and leak detection sensitivity.

- 8.6 Examination of failure statistics for other piping systems indicates that even where leak-before-break is automated, a significant proportion of failures occurring, do so without warning.
- 8.7 Leak-before-break is unlikely to be applicable to pressure tube failures initiating from fabrication flaws or hydride blisters.
- 8.8 A rejection index based on consideration of the following criteria has been recommended:
- limits related to design intent and normal operation
 - limits related to defect tolerance and:
 - limits related to property degradation.
- 8.9 Failure statistics and actual pressure tube experience indicates the two most likely causes of unexpected failure are deviations outside original design intent. This dictates that limits related to critical mechanical and operating systems shall be included in the rejection index.
- 8.10 The predicted ease with which delayed hydride cracking is likely to initiate at pre-existing defects under certain conditions suggests that maximum dimensional limits related to score marks, fret marks and other surface defects must be included. Cracks and lap type fabrication defects are not acceptable.
- 8.11 Deterioration of fracture properties of pressure tubes with time, neutron fluence, deuterium/hydrogen level etc. determines the need to set minimum critical crack length and maximum deuterium/hydrogen limits as part of the rejection index.
- 8.12 Consideration of presently required inspections and the recommended rejection index suggests that an expanded program of required inspections should include:
- (a) a pre-service inspection.
 - (b) a basic (or core) program of in-service inspection based on:

- monitoring of key mechanical and operating systems,
 - monitoring fracture properties for progressive deterioration due to neutron irradiation and deuterium pick-up,
 - continuous leak detection in the annulus gas system, and
 - sampling inspections to detect generic degradation mechanisms
- (c) an expanded sampling inspection to provide extra monitoring in the event that degradation, malfunctions or other deviations are detected during (a) and (b).

8.13 Consideration of the requirements presently specified in CSA N285.4 in the light of the present study, indicates that a major revision of that document is required to include the following features:

- (a) a revised philosophy regarding pressure tube inspection
- (b) inspection requirements based on the need to monitor deterioration in fracture toughness, defects and mechanical/operating system functioning.
- (c) a comprehensive set of guidelines on the evaluation of defects/deteriorations detected.
- (d) a set of acceptance criteria for known types of defects/deteriorations.

8.14 Review of background information available for this study suggested that a number of areas required further research by the nuclear industry. Specific recommendations made by the authors in this report are summarized in Appendix C. The most important are:

- (a) Establishment of reference delayed hydride cracking velocities for use in LBB analyses
- (b) Establishment of reference lower bound fracture toughness properties and critical crack sizes related to neutron fluence and deuterium levels and in particular to their combined effect.
- (c) More experimental work on initiation of delayed hydride cracking from possible surface defects to determine critical dimensions and stress intensities.

- (d) More detailed analyses of leak detection capability for each pressure tube in each operating reactor unit.
- (e) Greater understanding of oxidation and deuteriding kinetics in zirconium-niobium related to possible onset of breakaway hydriding, the effect of boiling and water chemistry in general.
- (f) More experimental work on the rate of growth of blisters, their relationship with bulk deuterium levels and the mechanism by which blisters nucleate delayed hydride cracking.
- (g) Further analysis of fatigue crack initiation at pre-existing surface defects and a reassessment of critical flaw sizes and the probability of fatigue being a likely mechanism of failure.
- (h) Obtaining more information on the properties of irradiated calandria tubes.
- (i) Consideration of the performance of pressure tubes containing defects under transient and upset operating conditions.

9.0 ACKNOWLEDGEMENTS

The authors gratefully acknowledge the contribution of Dr. Ahmed Shalabi, UNB, to this study. In particular, his advice on topics related to delayed hydride cracking was most appreciated.

APPENDIX A
BIBLIOGRAPHY OF INFORMATION



Analytical Approach to Fracture - General

G. J. Field. "Review of Developments During 1987 in Flaw Evaluation Procedures for Section XI of the ASME Codes" 826.222. February 1988.

E. M. Hackett et alia. "Fracture Resistance Characterization for LWR Materials" Origin unknown, 1988.

R. P. Harrison et alia. "Assessment of the Integrity of Structures Containing Defects", CEGB, R/H/R6 - Rev. 1. April 1977.

C. Michael Hudson et alia. "Case Histories Involving Fatigue & Fracture Mechanics" March 1985. ASTM STP918, 1986.

S. Kaiser. "A Deterministic and Probabilistic Model for Ductile Fracture" Int. J. Press. Ves & Piping, 22, 1-21, 1986.

Thomas P. Rich. "Case Studies in Fracture Mechanics" AMMRC MS 77-5. June 1977.

T. T. Shih. "An Evaluation of the Probabilistic Approach to Brittle Design" Engineers Fracture Mechanics, 13, 257-271, 1980.

L. A. Simpson. "Draft Report - Standard Test Method for Fracture Toughness of Candu Pressure Tubes" M&MB-88-244. June 1988.

ASME: "Acceptance Criteria for Flaws in Ferritic Piping and Appendix Z Evaluation of Flaws in Ferritic Piping" IWB-3650. December 1987.

"Development of Criteria for Assessment of Reactor Vessels with Low Upper Shelf Fracture Toughness" March 1987.

"Plane-Strain Fracture Toughness of Metallic Materials" ASTM E399-83, 1983.

"Standard Test Method for J_{IC} , A Measure of Fracture Toughness" ASTM E813-81, 1981.

"Engineering Approach for Elastic-Plastic Analysis" EPRI NP-1931. July 1981.

Blisters

D. V. Leemans and M. Leger: "Assessment of Blister Formation in the Pressure Tubes in the Point Lepreau Reactor" Ontario Hydro, 27-244-P, November 1987.

M. Leger et alia. "A study on Hydride Blister Formation in Bruce 3 and 4 and Pickering 5 and 6" Ontario Hydro, 86-253-K, November 1986.

M. P. Puls and W. R. Wallace: "Criterion for Crack Initiation from Zirconium Hydride Blisters" WNRE-710, May 1986.

A. Sawatzky et alia. "The Growth of Hydride Blisters in the Zr-2.5 wt% Nb Pressure tube WR1-12 (ZN)", WNRE-651, April 1985.

A. C. Wallace. "Experimental Investigation of the Cracking of Zirconium Hydride Blisters Grown on Zr-2.5% Nb Pressure Tube Material Trac Task 415/16.01 Progress Report 1" Ontario Hydro, 86-159K, July 1986.

Consequences of Failure

A. P. Muzumdar et alia.: "Experimental Program to Determine the Consequences of Pressure Tube Rupture in Candu Reactors" ANS. 1986.

A. P. Muzumdar et alia.: "Simulation of Candu Pressure Tube Rupture Experiments", ANS, 1986.

A. P. Muzumdar et alia. "Modelling of Full-Scale Pressure Tube Rupture Experiments", CNS, 1985.

V. J. Langman and . R. Weaver: "A summary of the assessment of fuel behaviour, fission product release and pressure tube integrity following a postulated large loss of coolant accident", AECB INFO-0114, May 1984.

Corrosion

B. D. Warr and P. C. Lichtenberger: "Corrosion Performance of Zr-2.5 wt% Nb Pressure Tubes in Ontario Hydro's Operating Candu Nuclear Reactors", Ontario Hydro, 1988.

D. O. Northwood et alia. "Corrosion and Hydriding Behaviour of Zr-2.25 Wt% Nb Alloy Nuclear Reactor Pressure Tubing" J. Materials for Energy Systems, 4, 1, 1982.

D. Lim et alia: "The Degredation of Zirconium Alloys in Nuclear Reactors - A review", AECSB, INFO:0174, 1986.

Delayed Hydride Cracking

R. W. Cahn. "Journal of Nuclear Materials" Volume 105, 1982.

B. A. Cheadle and C. E. Ells: "Crack Initiation in Cold-Worked Zr-2.5 WT% Nb by Delayed Hydrogen Cracking" AECL-5799. 1977.

C. E. Coleman et alia: "Minimizing Hydride Cracking in Zirconium Alloys" AECL, June 1985.

C. E. Coleman Memorandum: "Velocity of Delayed Hydride Cracking in the Axial Direction of Cold-Worked Zr-2.5 wt% Nb Pressure Tubes". January 1983.

A. B. Mitchell. "Pressure Tube Materials for Future Reactors" CNS-01202. March 1976.

M. P. Puls et alia: "Hydride-Induced Crack Growth in Zirconium Alloys" AECL-7392.

M. P. Puls et alia: "The Effect of Matrix Strength on Hydride Fracture in Zirconium alloys" CANDEV-86-94. December 1986.

G. K. shek. "Delayed Hydride Cracking Behaviour in Zr-2.5 Nb Under Simulated Depressurized Condition". Ontario Hydro, M86-131-K, December 1986.

L. A. Simpson et alia: "Fracture Toughness of Zirconium Hydride and its Influence on the Crack Resistance of Zirconium Alloys" AECL-6596, May 1979.

C. E. Coleman et al: "Control of microstructure to increase the tolerance of zirconium alloys to hydride cracking", AECL 9524, Dec. 1987.

L. E. J. Mooder: "Effects of hoop stress, operating temperature and surface defects on fracture of hydrided cold-worked Zr-2.5% niobium pressure tubes", CRNL 2695, May 1986.

Examination of PT's

Colin Angus. "Pressure Tubes...The Saga Continues", Ontario Hydro, Date Unknown.

C.D. Cann et alia. "An Electron Microscopical Examination of the X-14 Pressure Tube from Bruce Unit 2", WNRE-610, August 1984.

B.A. Cheadle et alia. "The Examination of the Pressure Tube From Pickering Unit 3 Fuel Channel J-09", CRNL-2597-10, January 1985.

B.A. Cheadle et alia. "The Examination of the Ruptured Zircaloy-2 Pressure Tube from Pickering NGS Unit 2 (Transcript of Video Presentation", AECL-8850, July 1985.

B.A. Cheadle et alia. "Examination of Core Components Removed from CANDU Reactors", AECL-9710. November 1988.

C.E. Coleman et alia. "Micro-Examination of the Crack in the Outlet End of the Pressure Tube Removed From Bruce NGS Unit 2 Fuel Channel NO6", CRNL-4044-03, May 1987.

B. Cox et alia. "Assessment of Oxidation and Deuteriding of P2 Tubes G-16 and J-15", CRNL-2597-04, February 1984.

P.H. Davies. "Interim Report on the Examination and Testing of the Zircaloy-2 Pressure Tube Removed From NPD Channel FO8", CRNL-4173-1, July 1987.

G.J. Field et alia. "The Failure of the Pressure Tube in Fuel Channel NO6 of Bruce NGS-A Unit 2 in March 1986", 2787a29, Date Unknown.

G.J. Field et alia. "Analysis of the Pressure Tube Failure at Pickering NGS "A" Unit 2 Nuclear Systems Department", AECL-9130, August 1984.

Figure 7.14: P1 P14 Pressure Tube Blister at 433.8cm from Inlet, CRNL-2597-11. June 1987.

E.F. Ibrahim. "Final Report on the Examination of the Pressure Tube Removed from Channel J-15 of Pickering Reactor 2", CRNL-2597-06, August 1984.

D.K. Rodgers et alia: "The Examination of the Pressure Tubes Removed from NPD Channels GO5 and GO7", CRNL-2597-13, January 1985.

D.K. Rodgers et alia: "Examination of the Pressure Tube Removed from Pickering NGS Unit 4 Fuel Channel K10", CRNL-4086-1, January 1987.

D.K. Rodgers et alia: "Examination of the Crack in the Inlet Rolled Joint of the Pressure Tube Removed From Pickering NGS Unit 3 Fuel Channel F13, CRNL-2928-1, October 1985.

D.K. Rodgers. "The Examination of the Cold Worked Zr-2.5wt% Nb Pressure Tube from Pickering Unit 4 Fuel Channel N16", CRNL-2597-15, August 1985.

D.K. Rodgers et alia. "Macro-Examination of the Crack in the Outlet End of the Pressure Tube Removed from Bruce NGS Unit 2 Fuel Channel NO6", CRNL-4044-1, July 1986.

D.K. Rodgers et alia. "The Examination of Components Removed from Pickering NGS Unit 1 Fuel Channels GO4, G14, G16, G20, K05, K18 and P14", CRNL-2597-11, June 1987.

A.R. Causey et alia: "The examination of the Pressure Tube Removed from Fuel Channel P12 or Bruce Unit 2", CRNL-4015, February 1987.

Fracture Analysis of PT's

B.A. Cheadle et alia. "CANDU-PHW Pressure Tubes: Their Manufacture, Inspection, and Properties" AECL-7564. December 8, 1981.

C.E. Coleman et alia. "Regression Analysis of Slit Burst Test Data for Cold-Worked Zr-2.5 wt% Nb Pressure Tubes" CRNL-2501, March 1983.

P.H. Davies et alia. "Flaw assessment of NPD Cold-worked Zircaloy-2 Pressure Tubes" CRNL-4046, September 1986.

P.H. Davies. "Energy Balance Approach to the Failure Analysis of Zr-2.5% Nb Pressure Tubes" 81-137-K, June, 1981.

C.E. Ells et alia. "The Pull-Out Strength of the Pressure Tube-End Fitting Rolled Joint in the CANDU Reactor". CRNL 2950, Feb. 1987.

Brian W. Leitch. Letter with enclosures to Professor E. Smith, Manchester, U.K. M&MB-87-181. October 27, 1987.

L.E.J. Mooder. "Flaw Tolerance of Hydrided Cold-Worked Zr-2.5 wt% Nb Pressure Tubes Prepared for Task Group III" CRNL-2452. May 1983.

M. Natesan et alia.: "An Assessment of Acceptable Crevice Corrosion Pits in Pressure Tubes", Ontario Hydro, 87340. Sept., 1987.

E.G. Price et alia. "Fast Fracture of a Zirconium Alloy Pressure Tube: Cause and Implications". AECL, Date Unknown.

L.A. Simpson et alia. "Standard Test Method for Fracture Toughness of CANDU Pressure Tubes" Draft Report. Date Unknown.

L.A. Simpson: "The use of J resistance curves to characterize the effects of microstructure strength and irradiation in fracture micromechanics in reactor pressure tubes", AECL 7954, 1983.

L.A. Simpson. "The Application of Ductile-Fracture Analysis to Predictions of Pressure-Tube Failure" AECL-6805, August 1981.

E. Smith. "Appraisal of Atomic Energy of Canada Limited Research Programs on Failure of Zirconium Alloy Pressure Tubes" WNRE-757, 1987.

B.J.S. Wilkins et alia. "Probabilistic Aspects of Fatigue Crack Propagation Data for Zirconium-2.5% Niobium", AECL-5529, November 1976.

Memorandum: Allowable flaw geometrics in irradiated Zr-2.5% Nb pressure tubes under SLAR and normal operating cyclic stresses, Ontario Hydro 87-292, 1988.

Fracture Properties of PT's

C.K. Chow et alia. "Fracture Toughness of the Burst Pressure Tube P2-G16" WNRE-637, CRNL-2597-09. March 1985.

C.K. Chow. Memorandum re: "Summaries of Fracture Toughness and Tensile Results of the Irradiated Pressure Tube Materials", MMB-88-425, December 22, 1988.

L.A. Simpson et alia. "Effect of Metallurgical Variables and Temperature on the Fracture Toughness of Zirconium Alloy Pressure Tubes", 939, 1987.

J.M. Smeltzer. Memorandum "Effect of Cooling Rate On Room Temperature Fracture Toughness of As-Received Zr-2.5 Nb Pressure Tube Material" MMB-88-153, 2-31-6536-110. October 5, 1988.

E.F. Ibrahim, "Mechanical Properties of Cold Drawn Zr-2.5Nb Pressure Tubes After Up To 12 Years in CANDU Reactors". Atomic energy of Canada Limited, Chalk River. Date Unknown.

L.A. Simpson. "The relationship Between Stress Intensity Factor, Crack Opening Displacement and J-Integral in ZR-2.5 Percent Nb", AECL 6725, January 1980.

G.D. Fearnough et alia. "Application of the Crack Opening Displacement approach to the Prediction of Pressurized Tube Failure", The International Journal of Fracture Mechanics, Vol. 4, Nr. 3. September 1968.

L.E.J. Mooder. "Effects of Hoop Stress, Operating Temperature and surface Defects on Fracture of Hydrided Cold-Worked Zr-2.5 wt% Nb Pressure Tubes", CRNL-2695. May 1984.

B.J.S. Wilkens. "Slit-Burst Testing of Cold-Worked Zr-2.5 Wt % Nb Pressure Tubing for Candu-PHW Reactors", AECL-6195.

C.K. Chow et alia.: "Fracture Toughness of Zr-2.5WT% Nb Pressure Tube from Pickering NGS-3 Fuel Channel J-09", CRNL2597-20, WNRE-712, May 1986.

E.F. Ibrahim et alia.: "Tensile Properties of Cold-Worked Zr2.5 wt% Nb Pressure Tubes Irradiated in Pickering-3 and -4 Reactors", CRNL-2414. August 1982.

General

D. J. Burns: "A review of safety issues related to the failure of the pressure tubes and fuel channels used in Ontario Hydro's Candu reactors". Ontario Nuclear Safety Review, Dec. 1987.

B. A. Cheadle, "Confidence in Fuel Channel Components" Date Unknown.

B. A. Cheadle et alia. "Fuel Channel Technology Seminar - Volume 1" CRNL-2950 OH D&D Gen 85431. February 1987.

B. A. Cheadle et alia. "Fuel Channel Technology Seminar - Volume 2" CRNL-2950 OH D&D Gen 85431. February 1987.

G. J. Field. "Overview of COG Fuel Channel R & D Program" Date Unknown. (1988?)

G. J. Field. "Fuel Channel Strategy and Program" COG/AECB Presentation, Feb. 1988.

Robert G. Hart. "Assessment of the Candu Pressure Tube Situation from the N.B.E.P.C. Perspective" January 1988.

R. J. Haslam. "A Brief on Candu, May 1988" TAD/R(88)1803. May 1988.

E. F. Ibrahim. "Development of Zirconium Alloys for Pressure Tubes in Candu Reactors" AECL-9125. June 1985.

V. J. Langman et alia. "A Summary of the Assessment of Fuel Behaviour, Fission Product Release and Pressure Tube Integrity Following a Postulated Large Loss-Of-Coolant Accident" INFO-0114. May 1984.

D. Lim. "The Degradation of Zirconium Alloys in Nuclear Reactors - A Review" INFO-0174. January 1986.

Dr. Derek O. Northwood. "The Production and Applications of Less Common Metals" Book 1 (Papers 2-29). November 1982.

E. G. Price. "Highlights of the Metallurgical Behaviour of Candu Pressure Tubes" AECL 8338, 1984.

Nicholas Teekman. "A Review of Nuclear Sources of Hydrogen Ions and The Effect of Hydrides on Candu Zirconium Niobium Pressure Tubes" Ontario Nuclear Safety Review September 1987.

"Minutes of Meeting on Probabilistic Approach to LBB COG.31". September 1988.

"Periodic Inspection of Candu Nuclear Power Plant Components". CAN3-N285.4-M83. December 1983.

Hydride Re-Orientation

M. Leger et alia: "Hydride Stress Orientation in Pressure Tube Materials Including Task group 3 Materials", Ontario Hydro, 83-282-K, August 1983.

In-Reactor Deformation

N. Badie et alia. "Creepsag of Zr-2.5% Nb Pressure tubes in Candu Reactors" 823.214. May 1988.

A. R. Causey et alia. "In-Reactor Deformation of Zr-2.5 wt% Nb Pressure Tubes" CRNL-4000. September 1986.

C. E. Ells et alia. "Predicted Creep Ductility of Zirconium Alloy Pressure Tubes in Power Reactors" AECL-9022. April 1986.

R. G. Fishell. "Bruce NGS-A Pressure Tube Elongation Study" 809.307. March 1986.

R. M. Griffin. "Periodic Inspection Report Fuel channel Pressure tubes Point Lepreau NGS Unit 1" IR-03550-12 Rev. 0. August 1987.

R. A. Holt et alia. "Model for the Spread of Contact Between Pressure Tube and Calandria Tube" Date Unknown.

R. A. Holt et alia. "The Effect of Temperature on the Irradiation Growth of Cold-Worked Zr-2.5 wt% Nb" 823.21. December 1988.

B. K. Kakaria. "Fuel Channel Elongation in 600 MWe Candu Reactors" March 1979.

J. D. Parker et alia. "Correlation of Pressure Tube Elongation with Materials Microstructure" 823.21. April 1986.

A. Salina-Rodriguez et alia. "Comparative Study of the deformation Behaviour of Zr-2.5 wt% Nb and Excel Pressure Tube Alloys" August 1984.

S. Unterbergs et alia. "Effect of Spacer Location and Operating Conditions on Contact Spreading" 823.21. Date Unknown.

A. B. Mitchell and H. C. Jamieson: CNS-1R-311-30, Creep Engineering Design Plan - Report of Task Group 1, Sept. 1978.

Inspection Techniques & Sensitivity

J. A. Baron. "Nondestructive and Fracture Evaluation Section 1987 Review and 1988 Work Program", Ontario Hydro, 88-38-K, July 1988.

M. P. Dolbey et alia. "Cigar Pressure Tube Ultrasonic Inspection Philosophy", Ontario Hydro, M85-116-K, January 1986.

M. P. Dolbey and A. L. Allen: "A Comparison of Pressure Tube Ultrasonic Inspection Sensitivities Cigar, Offcut Inspection and Manufacturing Inspection", Ontario Hydro, M86-66-K, June 1986.

B. Licht et alia. "Ultrasonic Inspection of Pressure Tubes F6 and N13 at Point Lepreau NGS" CRNL-2354, March 1982.

K. E. Lindenschmidt et alia. "Sizing Cracks in Candu Reactor Pressure Tubes Using Ultrasonic Crack Tip Diffraction", Ontario Hydro, 87-276-K, December 1987.

M. D. C. Moles et alia. "Ultrasonic Fast-Scan Blister Detection System", Ontario Hydro, 1987.

G. Van Drunen et alia. "Fuel Channel Inspection at Cordoba NGS" CRNL-2417, August 1982.

Leak Before Break - General

S. R. Doctor et alia. "Nondestructive Examination (NDE) Reliability for Inservice Inspection of Light Water Reactors" NUREG/CR-4469. March 1987.

P. Janzen. "A study of Piping Failures in US Nuclear Power Reactors" AECL-Misc.-204. April 1981.

B. Mukherjee et alia. "Darlington Leak Before Break Material Test Program J-Resistance Curves Volume 1 - General Summary, Discussion of Results and Conclusions", Ontario Hydro, OH87-232-K. 1987.

"Report of the US Nuclear Regulatory Commission Piping Review Committee" NUREG-1061. August 1984.

"Leak-Before-Break: International Policies and Supporting Research" NUREG/CP-0077. October 1985.

"Guide to the Periodic Inspection of Nuclear Reactor Steel Pressure Vessels" International Atomic Energy Agency, TR99, 1969.

Leak Before Break Candu-PTubes

B. A. Cheadle et alia. "Evaluation of Hydride Cracking in Zr-2.5 wt% Nb Pressure Tubes in Pickering Units 3 and 4". CRNL-2415., December 1982.

C. E. Coleman and L. A. Simpson. "Evaluation of a Leaking Crack in an Irradiated Candu Pressure Tube", AECL-9733, June 1988.

B. W. Leitch. "Theoretical Crack Driving Force and Leakage Area Calculations for Through-Wall Axial Cracks in a Zr-2.5 wt% Nb Pressure Tube" COG-87-67 RC-70. 1987.

E. G. Price et alia. "Leak Before Break Experience in Candu Reactors" AECL-9609, April 1988.

B. A. Shalaby et alia. "Leak Before Break and Leak detection Systems of Candu Fuel Channels" U80230/R June 1988.

L. A. Simpson. "Fitness-for-Service Criteria for Zr-2.5% Nb Pressure Tubes", WNRE-763, 1987.

B. J. S. Wilkins et alia. "Statistics of Leak-Before-Break", CANDEV-87-13. 1988.

WORLD EXPERIENCE

E.O. Adamov et alia. "Increasing the Safety of Nuclear Power Stations With RBMK reactors", Vol. 62, No. 4, Risley Trans 5492. April 1987.

A.P. Aleksandrov et alia. "Experience Accumulated by Soviet Nuclear Power Engineering", Nuclear Power in the USSR, UDC 621.039(47-57). April 1983.

Yu M. Cherkashov: "Safety Design of the RBMK-1000 Reactor", Risley Trans 5311. May 1983.

B.A. Cheadle et alia.: "Delayed Hydride Cracking in Russian RBMK ZR-2.5 NB Pressure Tubes", RC-87. August 1988.

G.C.W. Comley and D.J. Ferrett, "The contribution of Winfrith SGHWR in the development of water reactor chemistry technology", Nucl. Energy, 1981, Vol. 20, 391-402.

D.J. Ferrett et alia, "Winfrith SGHWR Operation Experience", 2nd International Topical Meeting on Nuclear Power Plant Thermal Hydraulics on Operations, Tokyo, April 1986.

M.D. Freshley et alia. "Investigation of the Combined Failure of a Pressure Tube and Defected Fuel Rod in PRTR", BNWL-272. May 13, 1966.

R.J. Haslam. "Pressure Tubes and Pressure-Tube Failures in Pressure-Tube Reactors", TAD/R(88)1805. May 5, 1988.

B. Negus and W. S. Black: "Winfrith SGHWR operating experiences 1968-1981" Nucl. Energy, 20, 5, 347-358, 1981.

"Summary report on the post-accident review meeting on the Chernobyl accident". IAEA, Vienna, 1986.

APPENDIX B

PRESSURE TUBE PERFORMANCE



APPENDIX B
PRESSURE TUBE PERFORMANCE
CATEGORY A DEGRADATIONS

A1 PRTR Hanford US 1965

Experimental test section of reactor failed during experiments involving the irradiation of deliberately defected fuel elements. Analysis of incident indicated that fuel melting had occurred. The Zircaloy-2 pressure tube did not fail at the fuel burst position because coolant was present. However, steam formation downstream of the fuel failure blanketed the pressure tube causing it to overheat and fail under pressure. The damage was a hole of 0.5 in diameter. The aluminum calandria tube remained largely undamaged.

There was no evidence to link any part of the failure to a deficiency in the pressure tube. Subsequent testing showed the pressure tube retained good fracture toughness and strength. The only degradation present was fuel fret marks up to 0.026 in. deep which had been known to be present in many pressure tubes since 1962.

A2 LUCENS: Switzerland: 1969

This carbon dioxide cooled, heavy water moderated reactor was being returned to power after a prolonged shutdown. At 50% power a complete depressurization of the coolant circuit occurred accompanied by fuel failure and loss of heavy water from the calandria. Detailed investigation carried out some ten years later attributed the failure to obstruction of the coolant flow through one fuel element leading to its melting, ignition and the rupture of the surrounding Zircaloy-2 pressure tube and calandria tube and the collapse of other calandria tubes. The damage was so extensive that the plant was decommissioned. Prior to the accident, faults with water cooled circulation pump seals had allowed water ingress into the coolant circuit and it was known that plugs of water could prevent coolant flow.

A3 Pickering Units 3 and 4, Ontario, 1974-5

Twenty tubes were removed from unit 3 and tubes from unit 4 as a result of leaks developing during early service of these units (see also B1). One leaking tube has been subsequently removed from unit 3. Analysis of the cracks causing these leaks determined the cause of failure to be due to delayed hydride cracking (DHC). Improper rolling procedures applied to the zirconium-niobium pressure tubes had resulted in high residual tensile stress in the region of cracking and this condition combined with an increase in deuterium content near the rolled joint and thermal cycling during commissioning and early operation, had provided all the prerequisites for this mechanism to occur.

A4 Pickering Unit 2, Ontario, 1983

A zircaloy-2 pressure tube, G16, failed suddenly while the reactor was at full power. This resulted in a severe loss of reactor coolant through the ruptured annulus sealing bellows. The reactor was shut down in a controlled fashion twenty minutes later. D₂O transfer and recovery systems were required but no special safety systems were required and no fuel failures occurred.

Failure was the result of a two metre long crack along the bottom of the pressure tube toward the outlet end. The final brittle pressure tube rupture had initiated at three small "blisters" located on the outside of the tube on the bottom. The blisters were approximately 0.160 in. in diameter and 0.040 in. deep. Final analysis speculated that individual delayed hydride cracks grew from each blister and finally linked to form a crack which exceeded the critical crack length for tube rupture. The deuterium levels found in the tube towards the outlet end were significantly higher than expected and had reduced the fracture toughness of the tube at operating temperature. The cause of the blistering was diagnosed as segregation of deuterium to cold spots on the pressure tube caused by contact between the pressure tube and calandria tube. Displacement of the garter springs during construction and/or commissioning was blamed for allowing this tube contact. The particular design of the garter spring was a major factor in its ineffectiveness.

A5 Bruce Unit 2: Ontario: 1982

This unit was in construction at the time of the Pickering Unit 3 rolled joint cracking. Examination showed the Bruce Unit 2 rolled joints to also be over-rolled. Rather than rebuild the unit, it was decided to reduce residual stresses in the rolled joints by individually stress relieving them. Despite this treatment, two rolled joints have since leaked due to DHC. It was suggested that the DHC may have initiated before the stress relieving treatment. The hydrogen/deuterium had increased to levels allowing propagation of cracks at operating temperatures.

A6 Bruce Unit 2: Ontario: 1982

A further rolled joint leak (J11) that developed in the same period as the leaks described above was attributed to DHC, nucleating at a fabrication flaw. The flaw consisted of a discontinuous lamination at a low angle to the inside surface of the tube. Although a positive leak indication was obtained, several months elapsed before this tube was identified as the source of leakage, indicating a weakness in the unit leak-before-break procedures.

A7 Bruce Unit 2, Ontario, 1986

Leakage from the zirconium-niobium pressure tube NO6 was detected during a hot poison-out and the reactor was cooled for leak identification. Whilst attempting this identification, the tube was pressurized at room temperature. This caused the tube to rupture in a brittle manner. The pressure tube rupture led to the failure of the calandria tube fuel damage and to the loss of fuel into the calandria.

Analysis of the failure which initiated in the region of the outlet rolled joint showed the presence of a very long, planar lamination at a shallow angle to the tube surface had been overlooked in fabrication inspection. This lamination had preferentially corroded, initiated DHC and eventually leaked. There were several DHC initiation sites in the rolled joint area and a 77 mm long crack had propagated. this length of crack was in excess of the critical crack length (estimated at 50 mm) when pressurized at ambient temperatures. The hydrogen/deuterium levels measured were such that no hydrides would have been present at outlet operating temperature, precluding DHC during normal operation. However, during the hot poison-out, there was sufficient hydrogen/deuterium to allow DHC to occur. It was after some hours holding at this intermediate temperature (200°C) that leakage was detected. Fractography confirmed that the majority of the DHC cracking had occurred in a continuous fashion at an elevated temperature, just prior to the rupture. Oxide wedging resultant from the corrosion of the defect was thought to have added to the stresses acting on the defect.

The lamination in the pressure tube extended through the rolled joint forming a "loose" flap of material. Fuel which had resided in contact with this flap, showed fretting damage. Examination of fuel defect records revealed a history of fuel damage in this channel extending back to 1980. It is not known if any examination had resulted from this poor fuel performance.

A8 RBMK Reactors, Russia, (up to) 1985

It has been reported that in the ten years operating experience with over 23,000 fuel channels, six have been removed because of leaks during operation. It is believed that the leaks occurred in the zirconium-niobium pressure tubes remote from the transition joints

at the ends of the pressure tubes. The leaks were attributed to DHC caused by very high residual stresses introduced by a straightening process used in their fabrication. The cracking problem was solved by changing the straightening process and increasing the final autoclaving treatment to 400°C to reduce residual stresses.

The RBMK reactors appear to be provided with a sensitive pressure tube leak detection system.

A9 Chernobyl-4, Russia, 1986

Physics experiments being carried out on the reactor core resulted in a sudden increase in reactivity and reactor power. The reactor reached 100 times full power in less than four seconds. Vapour explosions in many channels led to severing the reactor cover, rupture of all pressure tubes and further coolant voiding. A second power pulse and vapour explosion ejected one third of the fuel inventory, mostly downward, to terminate the excursion. Severe damage to the reactor and reactor building occurred. Prior degradation of the pressure tube is not suspected of playing any role in this incident.

CATEGORY B

B1 Pickering Units 3/4, Ontario, 1975

In addition to the leaking tubes reported in A3, thirty-nine additional tubes were removed from service because of rolled joint cracks detected by ultrasonic and eddy current inspection. These cracks were all attributed to DHC, induced by high stresses from incorrect rolling procedures.

B2 Bruce-2, Ontario, 1975

Fabrication flaw at 30° to pressure tube (H2) surface extended 40 mm through the rolled joint into the pressure tube was discovered by examining pressure tube off cut and later confirmed by eddy current testing. The tube was replaced.

B3 Cordoba, Argentina, 1982

An oxide filled lap, oriented partly at a low angle to the pressure tube (J21) surface and partly at right angles to the surface was detected during a preservice eddy current inspection. The tube was replaced.

B4 Pickering Units 1 and 2, Ontario, 1984

Careful consideration of the condition of the zircaloy-2 pressure tubes related to several types of deterioration led to the decision to retube these units (with zirconium-niobium tubes) a few years earlier than had been planned. It had been known for several years (results of monitoring) that limited bearing travel would lead to retubing due to pressure tube elongation. After the failure of tube P2 G16 due to blistering, intensive inspection programs confirmed both the presence of blisters in other tubes and that a large number of garter springs were displaced. Additionally, the tubes had considerably higher deuterium content and lower fracture toughness than anticipated.

B5 RBMK Reactors, Russia (up to) 1985

At least one zirconium-niobium pressure tube leaked during shutdown and was replaced. Cause of failure was as described in A8.

B6 Bruce-1, Ontario, 1986

Pressure tube L08 removed because of defect detected on inspection. Cause of defect is presently unknown.

B7 N-Reactor, USA, 1986

This reactor was taken out of service for safety considerations after the Chernobyl accident. Haslam reports that - "the condition of some of the pressure tubes is known to be questionable, sizeable defects having been found in some of the tubes examined".

B8 NPD, Ontario, 1986

A major consideration in the closure of the NPD reactor was the deterioration of fracture toughness in the 132 zircaloy-2 pressure tubes. Fracture toughness tests carried out on tubes removed for that purpose demonstrated that the critical crack length at room temperatures was about 20 mm and at operating pressures 70 mm. It was concluded that leak-before-break could not be assured at room temperature. Further, it was found that the pressure tube material remained in a non-ductile condition over the greater part of the operating cycle.

Although crack initiation was considered unlikely because of the generally good condition of the tubes, sufficient deuterium was found to be present along the full length of the pressure tubes to allow rapid propagation of DHC at operating temperature if it did initiate.

B9 Bruce-1, Ontario, 1988

Pressure tube removed (S04) because of pit like defects. Cause of this and above tube defect unknown. Suggestions for their origin include corrosion at inclusions or wear by debris.

CATEGORY C

C1 Pickering Unit 4, Ontario. (up to) 1986

Eight tubes with suspected small cracks in the rolled joint region have been left in service since 1974. Reinspections have failed to reveal any significant changes in length since then.

C2 Pickering Units 3 and 4, Ontario (up to) 1986

A number of defects/deteriorations effecting many pressure tubes are of particular concern:

- over-rolled joints in remaining original tubes
- unexpectedly high deuterium content of some tubes
- pressure tube elongation
- misplaced garter springs/pressure tube contact
- fuel fretting marks and
- lack of circulation in annulus gas system.

C3 Bruce Unit 2, Ontario. (up to) 1986

A number of defects/deteriorations effecting many pressure tubes are of particular concern:

- over-rolled but stress relieved joints
- pressure tube elongation
- displaced garter springs/pressure tube contact
- fuel fret marks and
- deuterium levels

C4 Bruce Units 1, 3 and 4, Ontario (up to) 1986

A number of deteriorations/defects effecting many pressure tubes are of particular concern:

- diametral creep (unit 3 only)
- pressure tube elongation
- displaced garter springs/pressure tube contact
- fuel fret marks and
- deuterium levels
- fuel coolant blockage from debris.

C5 Pickering Units 5 to 8, Ontario (up to) 1986

A number of deteriorations/defects effecting many pressure tubes are of particular concern:

- displaced garter springs/pressure tube contact (Units 5/6 only)
- pressure tube sag
- fuel fret marks and
- deuterium levels

C6 Bruce Units 5 to 8, Ontario (up to) 1986

A number of deteriorations/defects effecting many pressure tubes are of particular concern:

- pressure tube sag
- fuel fret marks and
- deuterium levels

C7 Wolsung, Cordoba, Pt. Lepreau (up to) 1986

A number of deteriorations/defects effecting many pressure tubes are of particular concern:

- displaced garter springs/pressure tube contact
- fuel fret marks
- pressure tube sag and
- different pressure tube chemistry due to boiling conditions may accelerate D₂ pick-up.

APPENDIX C

SUMMARY OF AUTHORS RECOMMENDATIONS FOR FURTHER WORK



C.1. Related to Section 2.1 Jurisdictional Requirements:

C.1(a) "It appears to the authors that to update N285.4 to become a credible working document on pressure tube inspection would require at least the following changes:

- (a) A statement regarding the philosophy of pressure tube inspection in terms of maintaining an acceptably low failure probability and in auditing generic deteriorations.
- (b) A statement regarding the designation of the present generation of pressure tube materials as "special" materials having more extensive periodic inspection requirements.
- (c) A recognition that due to neutron irradiation and/or deuterium pick-up, pressure tube fracture properties deteriorate in service. Monitoring of material properties is required to the same degree as defect monitoring.
- (d) A recognition that some failures and some potential failures can be avoided by correctly installed and operating fuel channel equipment. This also applies to the success of leak-before-break monitoring. Mechanical/operating inspections of key equipment must therefore be included.
- (e) A comprehensive set of guidelines as to how a wide range of presently known deteriorations and defects should be sized, evaluated and analyzed for acceptability.
- (f) A set of acceptance criteria for all known types for deteriorations.
- (g) An expanded inspection plan taking into account extra inspections when deficiencies are detected."

C.2. Related to Section 2.4 Leak-Before-Break Applied to Pressure Tubes

C.2(a) "DHC crack velocity appears to be dependent on the heating and cooling cycle, the operating temperature and the effect of neutron irradiation. These effects are shown in Figures 2-4. A prerequisite for cracking is, of course, that sufficient hydrogen/deuterium is present to form hydrides at the dwell temperature. In comparing the DHC cracking velocities with those quoted in the previously stated reports, it is noticed that the unirradiated velocity values are consistently used. If the apparent effect of irradiation is true (Figure 2) then significantly faster cracking rates should be used. Recalculation shows that in these circumstances, an LBB argument may only be sustained for short operating time periods or low fluences. This crucial point needs clarification."

C.2(b) "For most slow failure mechanisms such as fatigue and stress corrosion cracking, the crack propagation rates increase with stress intensity. This causes acceleration of cracking as the failure proceeds. For delayed hydride cracking, however, the presently available data (Figure 3) indicates it to be independent of stress intensity. A constant crack velocity is currently used in calculating the time interval from leakage to failure. This is also a crucial issue and the authors recommend further effort be expended to establish this independence from stress intensity beyond doubt."

C.2(c) "The determination of a completely acceptable lower bound relationship between CCL, fluence and hydrogen/deuterium effects, is essential in an LBB analysis and considerable effort should be directed toward this aim. It is not clear from the data reviewed as to the stress level used to calculate CCL. As discussed in Section 3, this stress level should match the most likely maximum stress identified in each reactor's safety report."

C.2(d) "The authors consider that the application of the LBB model given in Figure 1 to back-calculate the minimum acceptable critical crack length, would be a valid method for determining this rejection criterion."

C.2(e) "Another important consideration in the application of the LBB model is the likely profile of the cracks during growth. This has been discussed in the previously identified reports. In an idealized situation a length/depth ratio of four fits most observations, but in some instances the crack propagation may have been distorted by local stress fields at rolled joints to develop an aspect ratio as high as seven. This more pessimistic value should be assumed for calculation purposes."

C.2(f) "We had difficulty in establishing the operating duration for which LBB remained valid because of perceived uncertainties in critical metallurgical data. Further research effort is recommended to determine:

- an acceptable lower bound relationship between CCL, fluence and hydrogen/deuterium effects,
- the effect of flux and fluence on delayed hydride cracking velocity, and
- the influence of stress intensity on delayed hydride cracking velocity.

C.2(g) "The authors feel insufficient attention has been given to pressure tubes in several reactors which may be excluded from LBB monitoring because of annulus gas system monitoring deficiencies. A substitute inspection scenario for these tubes, and for the reactor as a whole if LBB becomes invalid in later life, needs to be developed. The overall effectiveness of LBB can only be determined if a detailed LBB analysis be carried out for each operating reactor unit and it is recommended that future studies be less generic and more specific to actual reactor units."

C.3. Related to Section 3.0 Pressure Tube Failure in Normal Operation, Upset and Accident Conditions

C.3(a) "Should the performance of a flawed pressure tube be analyzed under the worst anticipated operating conditions? Is it necessary to demonstrate that the pressure tube has a sufficiently long critical crack length to permit leak-before-break under the most pessimistic combination of high pressure and low temperature predicted?"

"A quick check on fracture analyses and predictions of critical crack length in generic studies, suggested that a nominal pressure (stress) is also generally used. The authors feel that for this reason they may give optimistic predictions of actual performance."

"The authors recommend that these concerns be addressed and a philosophy regarding operating transients be established."

C.4. Concerning Section 5.0 Potential Life Limiting Degradations

- C.4(a) "The authors concluded from the above that an auditing program designed to regularly monitor the rate and/or level of deuterium build-up along a number of pressure tubes and their rolled joints by scrape sampling, must be an essential feature of a future inspection program. An initial frequency of not greater than three years is thought prudent, since at the current time there are considerable uncertainties about the rate of build-up and the onset of breakaway conditions. Annual monitoring with an expanded sample base is suggested as appropriate in the period subsequent to breakaway if this, in fact, occurs. The selection of tubes for sampling should take into account compositional differences between tubes and the sample size should be sufficiently large to give confidence that representative levels have been monitored.

There are several possible deuterium/hydrogen levels that could be specified as the basis of rejection, depending on specific circumstances. The implication of each one of these will be discussed in subsequent sections. A selection is given in Table 1. A general rejection limit of 100 ppm is recommended for pressure tubes in defect free channels installed with low stress rolling procedures and with correctly positioned garter springs. The arbitrary level was chosen as a conservative maximum level after which further service could only be justified on the basis of detailed review. In some circumstances, as described in Table I and Sections 5.4 to 5.12, lower limits may have to be set to compensate for installation or other deficiencies."

- C.4(b) "Periodic monitoring of equivalent hydrogen levels of selected rolled joints will give adequate warning of excessively rapid build-up and of unacceptable levels of deuterium. Regions of the pressure tube just inboard of the rolled joint which are more likely to lead to a loss of strength in the event of excessive hydriding should be specially monitored. In the absence of any guidelines in other literature, an arbitrary limit of 200 ppm equivalent hydrogen is recommended by the authors as a conservative maximum level that can be tolerated without concern about reduced

pull out strength. This limit could possibly be increased in the future in the light of more experimental information."

C.4(c) "There are several possible approaches to setting a minimum fracture toughness limit. However, it is recommended by the authors that, initially, an LBB scenario be a requirement for pressure tubes and that minimum fracture toughness requirements be set with this in mind. Consequently, the basis of setting the limit should be the minimum critical crack length for any pressure tube in a particular reactor, that is required to fulfil the LBB detection scenario described in Section 2.4. Specific leak detection sensitivities and other operating conditions related to the particular reactor unit should be applied in this calculation."

C.4(d) "The authors recommend that a "minimum critical crack length" rather than a minimum "fracture toughness" be the requirement since this is both directly related to the LBB model and takes into account the variables of stress, fracture toughness and temperature, applicable to different locations in the pressure tube and rolled joints."

C.4(e) "It is recommended by the authors that an interim reference set of fracture toughness/critical crack length properties be established, as representing lower bound values. All pressure tubes should be judged against these curves, on the basis of their known neutron fluence, operating conditions and the results of scrape samples (D_2 analysis) taken directly from them. In the event that the calculated critical crack lengths approach the LBB limit, actual fracture properties should be measured on sample pressure tubes removed from the reactor for that purpose."

C.4(f) "The authors feel that a limit to the acceptable depth of fret marks, score marks and other surface imperfections is required, but do not have available the necessary information to suggest limiting values. It is recommended that more analysis and experimental studies be done on this topic."

C.4(g) "One possible way to eliminate DHC completely as a potential failure mechanism occurring during reactor operation is to limit deuterium/hydrogen content along the pressure tube to below the hot solubility limits. These values are likely to be exceeded after only a few years service and if applied as a limit could significantly shorten allowable pressure tube life. Consequently, the authors recommend an alternate approach based on: establishing an LBB scenario; sensitive fabrication inspection to ensure the absence of lap type defects, installation inspection to indirectly demonstrate rolled joints of low residual stress; and an in-service inspection of a sample of tubes to audit the dimensions of typical surface degradations that may occur during service."

C.4(h) "The high likelihood of multiple blister formation, low-aspect cracks and rapid delayed hydride cracking rates at operating temperature all suggest that LBB is unlikely to apply to tubes failing by this mechanism. Consequently, if pressure tube contact cannot be avoided (i.e. displaced garter springs), limiting acceptable deuterium/hydrogen levels to below the appropriate threshold value (see above) is recommended by the authors as a control measure to prevent these failures. Periodic inspection of a sample of pressure tubes to check for the development of blisters is also recommended."

C.4(i) "In reviewing this information, the authors concluded that: failures initiated by blisters are unlikely to be detected by "leak before break" and that for reactors with displaced garter springs, both scrape sampling for deuterium level and sample volumetric inspections to detect blister formation are essential. Recommended rejection limits for blistering are: displaced garter springs; and the presence of any blister. At some later stage, it would seem likely that a maximum blister size based on diameter, depth and proximity of blisters may become acceptable."

C.4(j) "The principal limit for elongation is allowable bearing travel. To maximize this travel, tubes are allowed to elongate at one direction and then the other. It is recommended that an initial limit be set at 75% of the available bearing travel in each direction. Limits as high as 90% could be allowed if each individual channel

is measured. A secondary limit needs to be applied to feeder spacing. This spacing must be maintained to prevent feeder fretting. Differential elongation between adjacent fuel channels could conceivably reduce the feeder spacing and a minimum gap of 6 mm is recommended."

C.4(k) "A recent study has established that the likely rupture ductility for zirconium-niobium pressure tubes is greater than 5% diametral expansion. This value is likely to be conservative with respect to the other two criteria for most reactors and is therefore recommended as the diametral creep limit. An exception to this may be Bruce A Unit 3 where garter spring arrangements may lead to the latter limit (garter spring lock-up) being exceeded."

C.4(l) "Further work is still required to determine fatigue properties and define acceptable lengths/depth dimensions for typical surface defects."

C.4(m) "Both major mechanisms of pressure tube failure; rolled joint cracking and blistering, can be attributed to deficiencies introduced during design and/or construction. Incorrect operation and/or poor flow conditions in annulus gas systems are suspected of increasing deuterium pick-up and preventing sufficiently sensitive leak detection capabilities. Optimum settings of bearing travel, feeder clearances and other mechanical equipment is required if pressure tube distortions are to be accommodated without creating other problems.

The authors recommend that periodic auditing of these critical system parameters be recognized as an important requirement and be included in the periodic inspection program. Deficiencies detected should require detailed investigation and it is also recommended that additional inspections for blistering, deuterium build-up, dimensional interference, fracture property degradation, etc., caused by these deficiencies, be implemented."